December 1957

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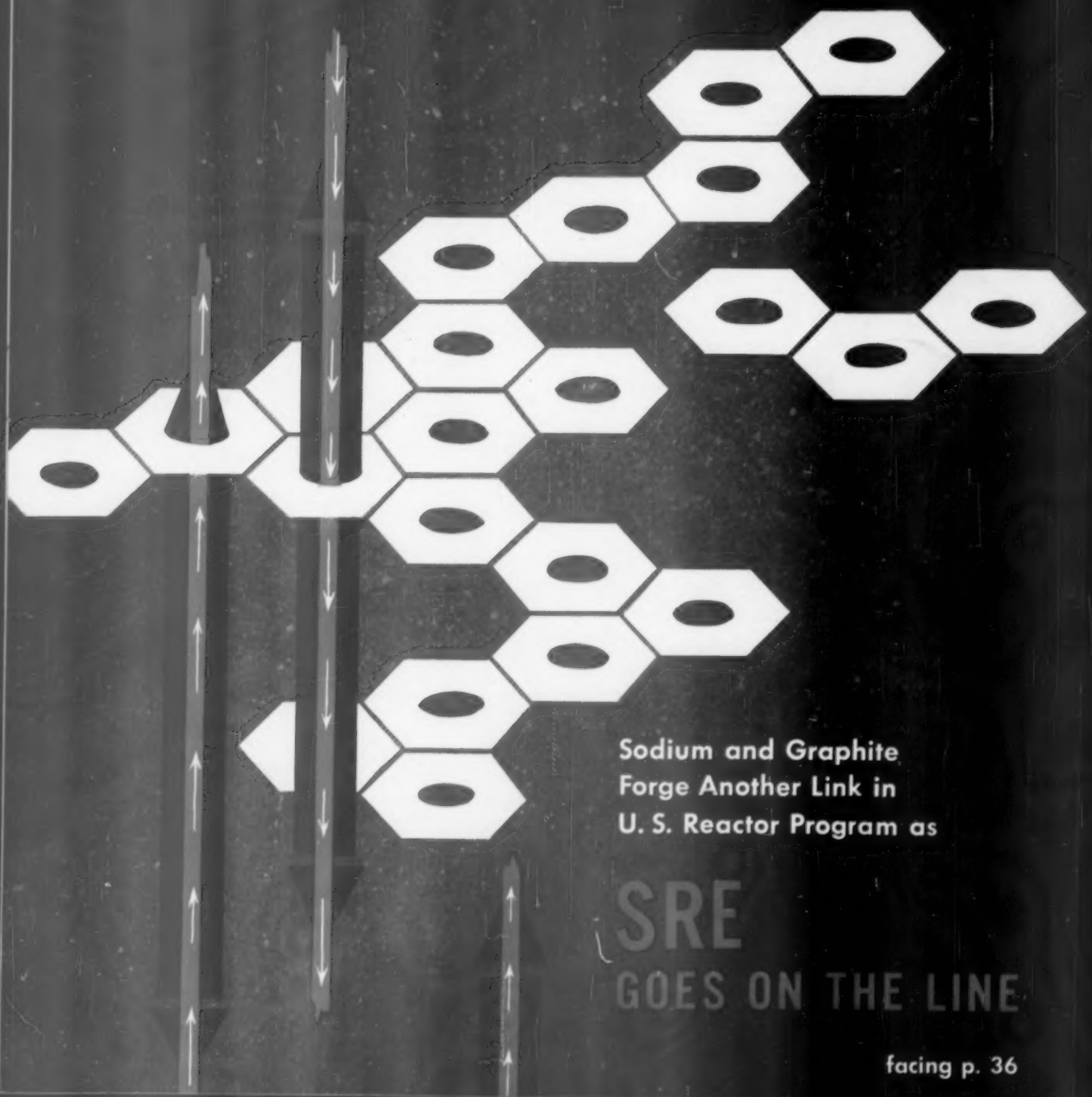
NUCLEONICS

ATOMIC POWER • NUCLEAR ENGINEERING • APPLIED RADIATION



Sodium and Graphite Forge Another Link in U. S. Reactor Program as

SRE GOES ON THE LINE

facing p. 36

ENGINEERING LEADERSHIP
KEY TO ATOMIC PROGRESS

General Electric engineers combine high power and flexible experimental space in new test reactor

A major advance in reactor technology was made by General Electric engineers and scientists in designing and developing the new 30,000-kw General Electric Test Reactor (GETR).

A new concept in test and research reactors, the GETR permits a wide variety of simultaneous high-flux experiments to meet the needs of a diversified research program. The advantages of high specific power in a tank-type reactor and the large, easily accessible experimental space of a pool-type reactor are combined in the GETR.

The first GETR will be installed at the General Electric Vallecitos Atomic Laboratory. General Electric will use the reactor for fuel development and general research programs. Experimental space will be available for other organizations within the industry. The design and development of this new reactor is another example of General Electric engineering leadership in the field of atomic energy. Atomic Power Equipment Department, General Electric Co., San Jose, California.

196-9

Progress Is Our Most Important Product

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NUCLEONICS

ATOMIC POWER
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APPLIED RADIATION

VOLUME 15 NUMBER 12

DECEMBER 1957

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About NUCLEONICS

We're Honored

Within the cornerstone of the new AEC headquarters building in Germantown, Maryland, is a metal box containing mementos of this atomic age. The mementos range broadly: from motion pictures of President Eisenhower's 1953 Atoms-for-Peace address before the United Nations to radiocarbon-dated linen wrappings from the Dead Sea scrolls. All the major AEC service awards are included as are specimens of atomic energy materials from raw uranium ore to finished fuel elements.

Included among the mementos, we are pleased to be able to report, is a microfilm copy of the first issue and the Tenth Anniversary Issue of NUCLEONICS. We are honored by this recognition of the part our magazine has been privileged to play in telling the story of these hectic yet wonderful ten years.

Index Time

Metal boxes sealed in cornerstones are one way of passing information along to future generations—a well-indexed scientific literature is another. An index is, after all, the key to unlocking the information stored in the issues it covers. It is for this reason that we pay particular attention to the completeness and utility of the index that is bound into the December issue each year.

This year being no exception we invite you to turn to page 116 and peruse the wares there displayed. A few minutes spent now getting to know the categories used will help in seeking out information later. Also you may come across an article title you missed and want to catch up on.

Nuclear Accidents

Everyone in this business is keenly aware of the need to do everything possible to insure the safe operation of nuclear facilities. Nevertheless accidents will occur, and when they do they provide lessons of prime significance to all who deal with the atom.

It is for that reason that we made space available in this issue for the text of official reports on two recent nuclear accidents—that at Britain's Windscale production reactor (p. 43) and that at the M. W. Kellogg Co.'s Houston, Texas, isotope lab last March (p. 42).

We plan to make a regular feature of these official accident reports and come out with them as soon after they appear as is possible. We are sure you join us in the hope that we will not be publishing many of them.

Fusion

Everywhere we go we see evidence of the very real interest in thermonuclear power. During Nuclear Week in New York it was the large crowd packed into a hot lecture hall to listen to Marshall Rosenbluth talk on the pinch effect. Last month it was Winston Bostick of Stevens Institute of Technology lecturing to a record turnout of the Princeton IRE section on "What We Can Learn from the Russians about Controlled Thermonuclear Fusion." We mean to report the fusion development, to you as they occur. For our views on classification policies in this area see this month's editorial (p. 35).

—The Editors

MICROFILM COPIES OF NUCLEONICS Tenth Anniversary Issue (September, 1957) and very first issue (Vol. 1, No. 1, September, 1947) were included (arrow) among mementos of the atomic age placed in cornerstone of new AEC headquarters building. Other contents include MTR fuel element section (upper right), uranium fuel slug (far right) and pencil-type dosimeter (bottom right)

NUCLEONICS (with which are consolidated "Atomic Power" and "Atomic Engineering"), December, 1957, Vol. 15, No. 12. Published monthly by the McGraw-Hill Publishing Co., Inc., James H. McGraw (1860–1948), founder. Executive, Editorial, Circulating and Advertising Offices, McGraw-Hill Building, 330 W. 42nd Street, N. Y. 36, N. Y. Publication Office, 212 East York St., York, Pa. See panel below for directions regarding subscriptions or changes of address. Donald C. McGraw, President; Joseph A. Gerardi, Executive Vice President; L. Keith Goodrich, Vice President and Treasurer; John J. Cooke, Secretary; Nelson Bond, Executive Vice President, Publication Division; Ralph B. Smith, Vice President and Editorial Director; Joseph H. Allen, Vice President and Director of Advertising Sales; A. R. Venezian, Vice President and Circulation Coordinator. United States Subscription rate for individuals in the field of the publication $8 per year; Canada, other Western Hemisphere countries, and the Philippines, $10 for 1 year; all other countries, $15 for 1 year. Single copy prices: United States, United States possessions, and Canada, $1; elsewhere, $2. Position and company connection must be indicated on subscription orders. Second-class mail privileges authorized at York, Pa. Printed in U. S. A. Title registered in U. S. Patent Office. © Copyright 1957 McGraw-Hill Publishing Co., all rights reserved.

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Coming Features in NUCLEONICS

Next Month—Nuclear Fuel Economics

Electrolytic Uranium

Reactor Siting

Later Months—Fusion Reactor

Special Report on Control Systems

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Send for Technical Bulletin ... also folder illustrating Blickman-Built Stainless Steel Specialized Laboratory Equipment.

LOS ALOMOS TYPE E/5

These Blickman-Built stainless steel vacuum rack enclosures are designed for the safe handling of radioactive materials in research applications. Two types of racks are available.

TYPE E5-1 has a vertically-rising door. A similar door can be provided on the opposite side if operation from two sides is desired.

TYPE E5-2 has sliding doors with can also be provided on the opposite side for two-sided operation. Available in various lengths to meet specific requirements.

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Seamless, crevice-free, round-corner construction assures easy cleaning, facilitates decontamination.

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If you are concerned with:

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Atomic energy projects...
Nuclear problems...

This data book will help you

This new free Norton *"Handbook on Boron Carbide and Elemental Boron"*, is a valuable reference, containing both fundamental data and practical information. Printed in colors with many charts and tables it will serve as a useful and permanent addition to the reference files of those concerned with atomic energy and related fields.

In nuclear science and engineering the importance of the neutron absorption reaction of the boron isotope B^{10} is well recognized.

Now, for the first time, available data on the development of numerous stable, boron-rich materials have been compiled and coordinated by Norton in one convenient reference book.

This book is the *"Handbook on Boron Carbide and Elemental Boron"* — a compact source of technical information on materials made by Norton for use in the atomic energy field.

A partial list of the table of contents includes: *Boron for neutron absorption . . . Technical grade boron . . . Properties of boron . . . Commercial grades of boron carbide . . . Radiation damage to boron carbide . . . Bonded boron carbide (carbon, silicate, plastic, metal) . . . Boron nitride . . . Metal borides.* These and many other subjects are fully described and illustrated with tables, charts and photographs.

A Quarter-Century of Experience

Work in Norton laboratories on boron compounds began some 25 years ago. This included the development of NORBIDE® boron carbide, the hardest material yet produced commercially. Norton facilities for producing NORBIDE wear-resistant articles and abrasive have been expanded to supply boron carbide in numerous forms for atomic energy applications.

Recent Norton development has been aimed at improving the quality, increasing the production and reducing the costs of various boron-rich products. As a result, Norton now produces these materials to highest purity standards, at prices ranging between one-half and one-tenth of former pricing.

Other Norton Electric Furnace Products

of special interest to nuclear engineers include ALUNDUM® fused alumina, CRYSTOLON® silicon carbide, MAGNORITE® magnesium oxide, FUSED ZIRCONIA and various refractory carbides, oxides and nitrides.

Besides being the basic ingredients of the famous Norton Refractory Rx's — refractories *engineered and prescribed* for the widest range of conventional applications — these high-melting materials are finding many new and valuable uses in atomic energy projects.

They are all described in the new Norton handbook. For your free copy, write to NORTON COMPANY, Refractories Division, 661 New Bond St., Worcester 6, Massachusetts.

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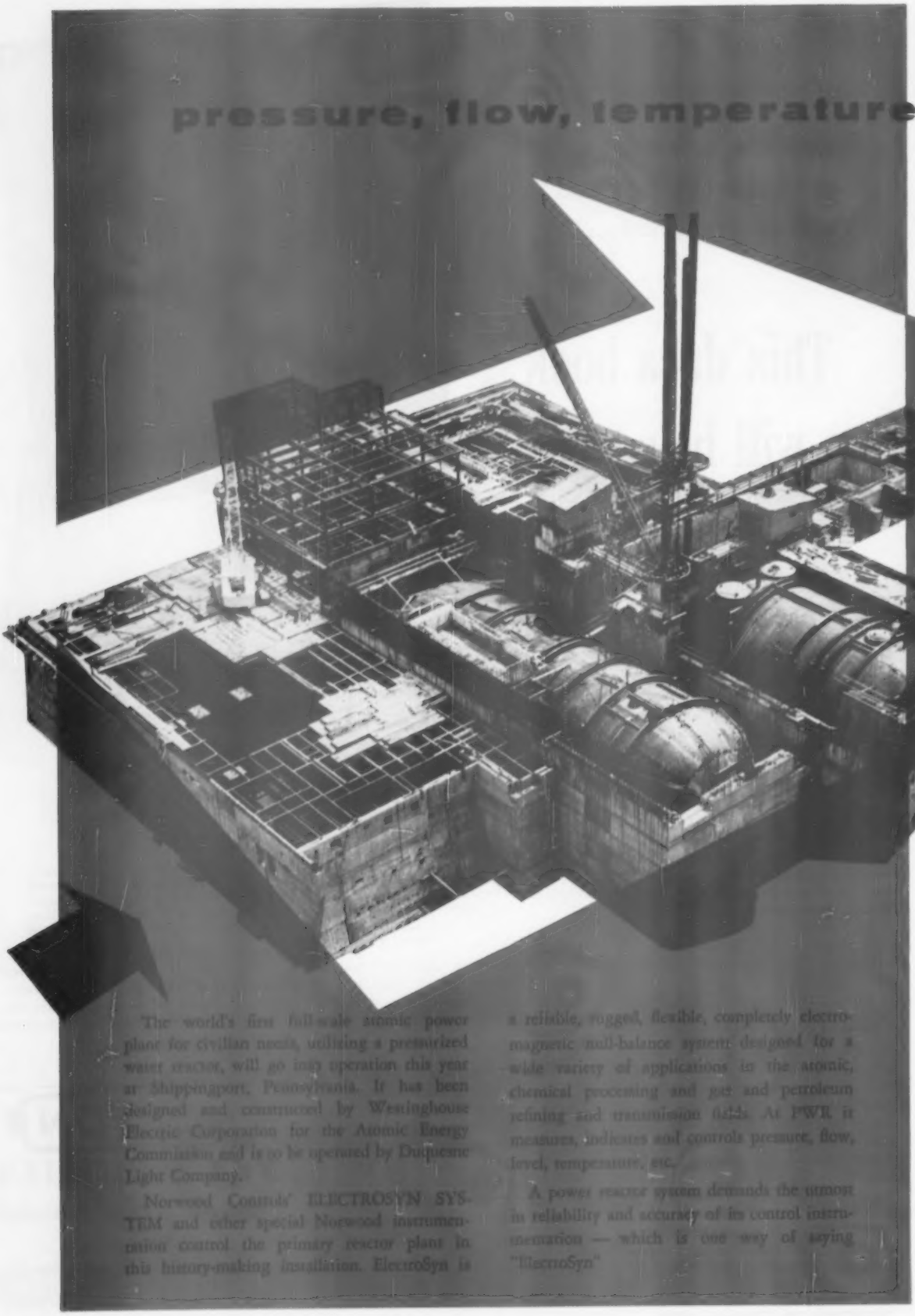
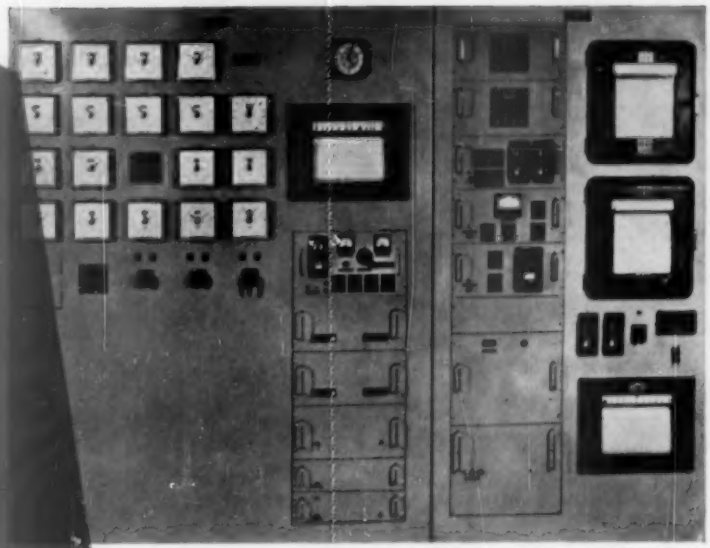


The world's first full-scale atomic power plant for civilian needs, utilizing a pressurized water reactor, will go into operation this year at Shippingport, Pennsylvania. It has been designed and constructed by Westinghouse Electric Corporation for the Atomic Energy Commission and is to be operated by Duquesne Light Company.

Norwood Controls' ELECTROSYN SYSTEM and other special Norwood instrumentation control the primary reactor plant in this history-making installation. ElectroSyn is a reliable, rugged, flexible, completely electromagnetic null-balance system designed for a wide variety of applications in the atomic, chemical processing and gas and petroleum refining and transmission fields. At PWR it measures, indicates and controls pressure, flow, level, temperature, etc.

A power reactor system demands the utmost in reliability and accuracy of its control instrumentation — which is one way of saying "ElectroSyn"

in PWR, FIRST U. S. full-scale atomic power station . . . controlled by ELECTROSYN SYSTEM



Norwood Controls' Power and Temperature Control Panel and Auxiliary Instrument Panel (at left) are the heart of the PWR reactor plant instrumentation and control system. Measurement, computation, auctioneering, alarm, and control functions are performed.

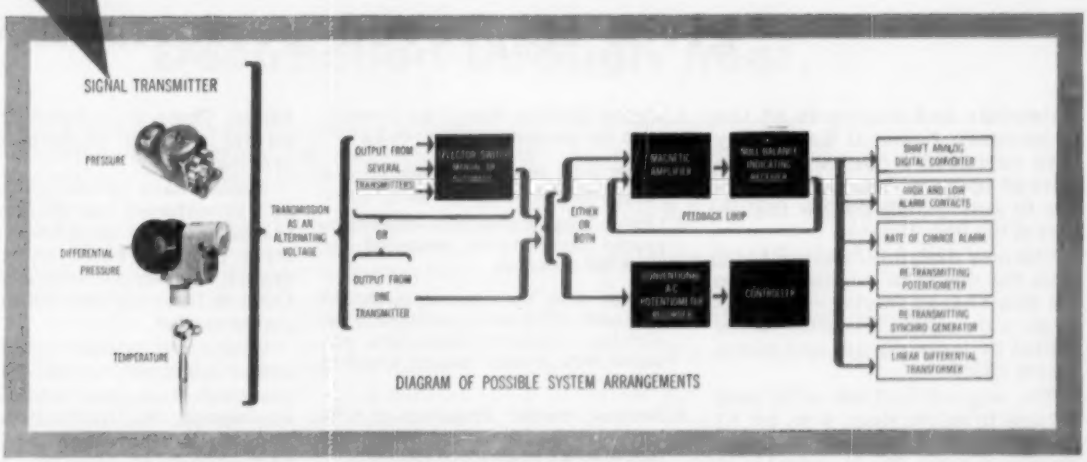


DIAGRAM OF POSSIBLE SYSTEM ARRANGEMENTS

Write for descriptive literature on "ELECTROSYN SYSTEMS"

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NC NORWOOD CONTROLS

Loading Face of BNL Research Reactor at
Brookhaven National Laboratory, Upton, N. Y.

New Brookhaven fuel element gives improved performance

Scientists and engineers of the Brookhaven National Laboratory have conceived a reactor fuel element of advanced design that promises to greatly improve the usefulness of the BNL Reactor.

The new design differs markedly from the one used originally when the Brookhaven reactor — the first large-scale research reactor in the United States—was put into operation in 1950.

The original fuel elements were natural uranium slugs, 4 in. by 1.1 in. in diameter, canned in finned aluminum tubes. The improved two-foot element, shown above, is an assembly of three fuel plates which contain highly enriched (93%) uranium-aluminum alloy clad in aluminum. The new element is expected to afford five important advantages:

1. Higher neutron flux — an increase from the present 5×10^{12} up to 3×10^{13}.
2. Lower total reactor heat output — a reduction from 24 to 20 megawatts.
3. Less cooling air required—due to improved aerodynamic design of the enriched elements.
4. Higher fuel burn-up of contained U^{235}-up to 40% burn-up with enriched elements compared with 10% obtained with present natural uranium fuel.
5. Reduced reactor inventory of U^{235}-from 385 Kg contained in present fuel to 55 Kg in enriched elements.

With more than eight years' pioneering experience in the fabrication and development of all types of fuel elements, Sylvania-Corning Nuclear Corp. was chosen to supply several hundred test elements of the new design. These were installed in the central "hot spot" of the reactor for proving out.

Results have proven satisfactory and Brookhaven has decided to go ahead with the complete replacement program based on the new design. Sylvania-Corning Nuclear Corp. will supply the 5000 fuel elements needed.

Whether your own reactor is in actual operation, or merely in the planning stage, our scientific and engineering staff is ready to assist you, too, in achieving improved fuel performance. For your files, write to us at Bayside, N. Y. for our booklet on Nuclear Fuel Elements.

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COBALT 60 PRICES REDUCED UP TO 70%

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NUCLEAR SYSTEMS

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Budd

Vallecitos Atomic-Electric Power Plant...

ON THE LINE

Nation's first privately financed atomic power plant begins operation

On October 24, 1957, this country's first all privately financed atomic-electric power plant began feeding 5,000 kw of electricity into the Pacific Gas and Electric Company's transmission network. The General Electric boiling water reactor which furnishes steam to a conventional turbine-generator, owned and operated by PG&E, was issued power reactor License No. 1 by the Atomic Energy Commission.

Operation of the reactor has been successful: the system is stable, responds quickly to load changes, and steam capacity is greater than presently licensed limits. Low radioactivity level permits direct maintenance of the turbine during operation.

Located at the General Electric Vallecitos Atomic Laboratory, the reactor is now providing operational data for the 180,000-kw Dresden Nuclear Power Station which General Electric is building for Commonwealth Edison Company and the Nuclear Power Group, Inc.

Designed and built by General Electric in co-operation with Pacific Gas and Electric, a member of the Nuclear Power Group, the Vallecitos Atomic-Electric Power Plant is a tangible step forward in the development of economic commercial nuclear power. Atomic Power Equipment Department, General Electric Company, San Jose, California. 197-6

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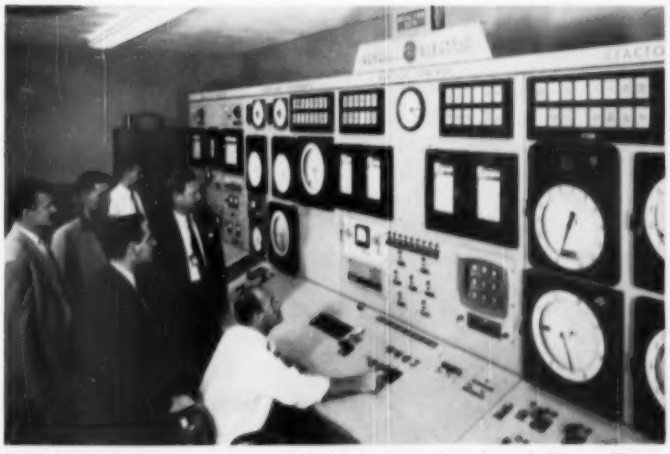

5000 kw was generated Oct. 24, 1957, at Vallecitos Atomic Power Plant, 17 months after ground-breaking. The General Electric reactor, built at a cost below original estimate, operates on AEC power reactor License No. 1.

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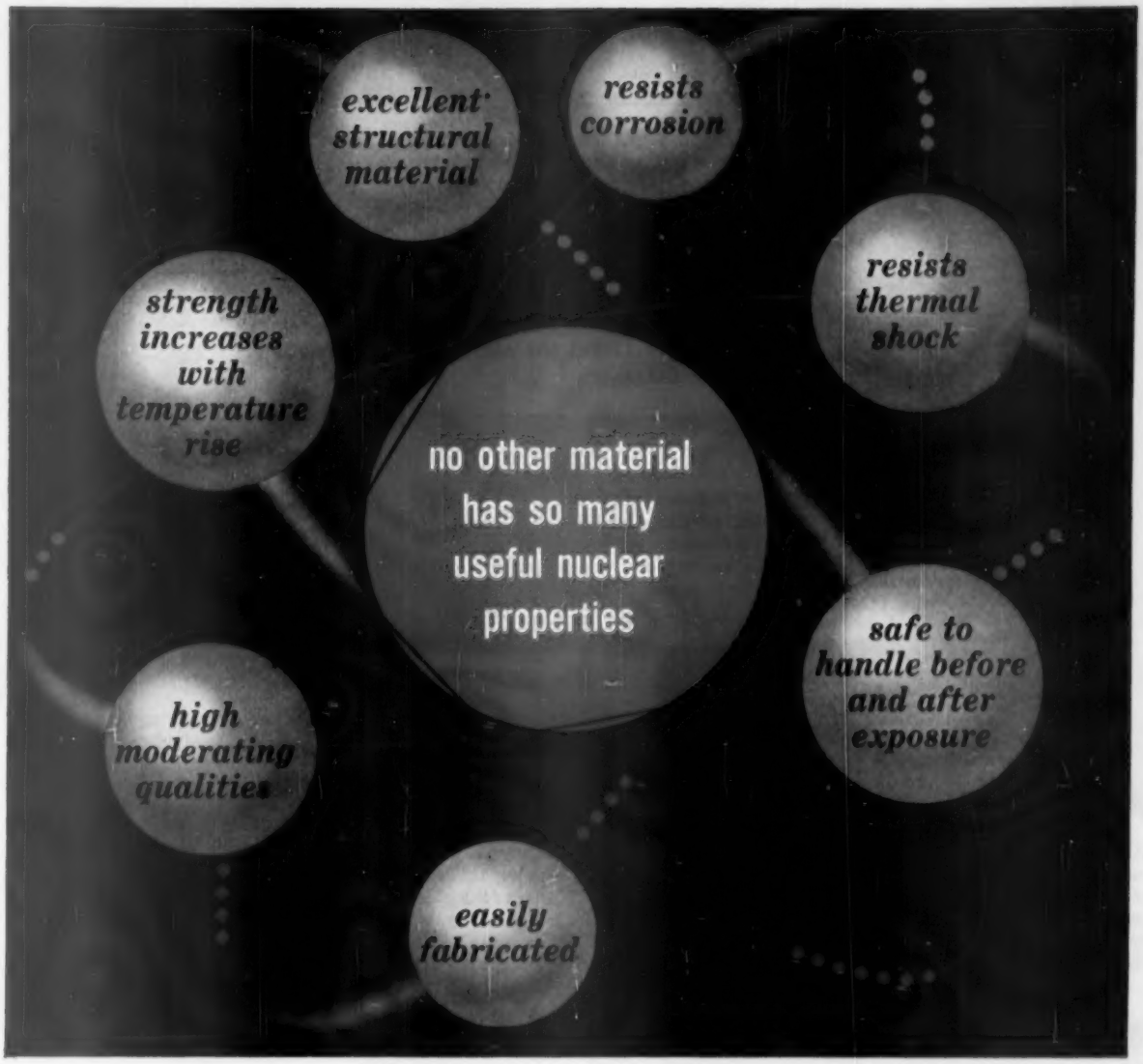
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"National" Nuclear Graphite was used in early reactors because of its excellent moderating properties and low neutron cross-section. Its low cost, safety, easy machinability and high refractory qualities also combine to make it an indispensable structural material. *Moreover, the addition of boron in low percentages makes graphite excellent for shielding.*

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rigid specifications

Tension tests are required to be made at room temperatures and at 670° F. The following minimum physical properties shall be met:

At Room Temperature:

TS	YS	EL	RA	CHARPY V-NOTCH
70,000	30,000	45	50	50

At 670° F. the minimum tensile strength shall be 51,000 p.s.i. and the minimum yield strength 18,300 p.s.i.

Rejection

Each casting that develops unacceptable defects during shop working or fails to conform to all of the requirements of these specifications shall be rejected. No repair by welding or other means will be permitted.

Radiographic Inspection

(a) Paragraph S5 (a) of the Supplementary requirements of ASTM-A 362-52T.

(b) All castings shall be radiographed 100% and shall conform to ASTM-E7 1-52, Class 2 quality, except as modified by these specifications.

(c) The manufacturer shall establish a positive system of identification of the X-ray plates which shall be subject to approval by the inspector. This system shall guarantee complete coverage by radiographing and provide for positive identification between the plate and the subject.

All cast pipe shall be hydrostatically tested to 5,900 p.s.i. and held at that pressure for 20 minutes with zero pipe leakage. Each length of pipe shall be hydrostatically tested at the manufacturer's plant.

Inspection of Penetrants

All castings shall be subjected to inspection by fluorescent penetrants or penetrating dyes both inside and out. All cracks, porosity, or flaws revealed as a result of the Dye Penetrant Test shall be due cause for rejection of the casting.

The 304L stainless steel shall conform to the following ladle analysis:

Carbon	.03 max.		
Manganese	1.50% max.		
Phosphorous	.03% max.		
Sulphur	.03% max.		
Silicon	2.00% max.		
Chromium	18.00	-	21.00%
Nickel	8.00	-	11.00%

Pipe: All pipe of the following sizes shall be centrifugally cast stainless steel as per ASTM-A 362-52T, except as modified by these specifications:

16" - Sch. #160
12" - Sch. #160
10" - Sch. #160
8" - Sch. #140

All pipe shall be machine finished to 125 micro-inch interior and exterior.

for nuclear power met by U.S. PIPE patented process

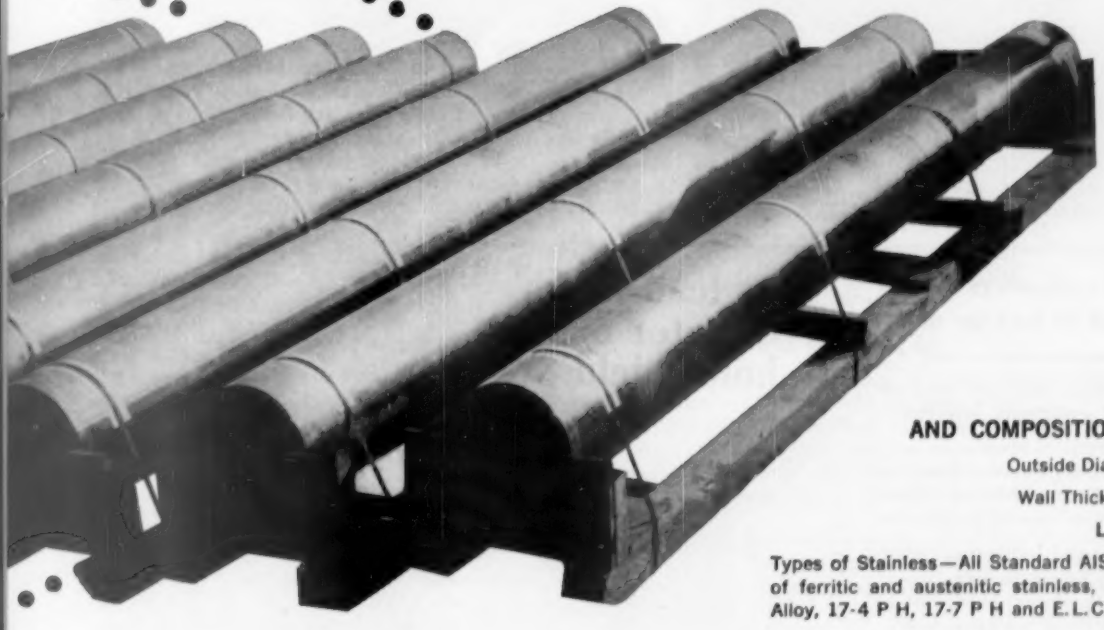
Centrifugally Cast Stainless Steel Solves Many Piping Problems

Combinations of temperatures, pressures and corrosive conditions never encountered before: these are among the piping problems that must be overcome by the men who design the nation's nuclear power installations.

Stainless steel centrifugally cast pipe provides many of the answers. Study the specifications at the left . . . specifications demanded of stainless steel pipe on a recent job for Paul Hardeman, Inc., Los Angeles, California. This pipe is being used for heavy duty, high pressure, elevated temperature service in the primary piping system of the SPERT-III Reactor at the U. S. Atomic Energy Commission's National Reactor Testing Station near Idaho Falls, Idaho. The Stearns-Roger Mfg. Company, Denver, Colorado, is the architect-engineer on this project. A complete tabulation of the actual test data obtained on this pipe and to this specification is available upon request.

U. S. Pipe is headquarters for metal mold centrifugally cast alloy and stainless steel pressure pipe over a wide range of special and standard analyses—in large and small quantities—and to individual specifications.

If piping of the type described above is the bottleneck in your nuclear power planning, write and outline the problem.



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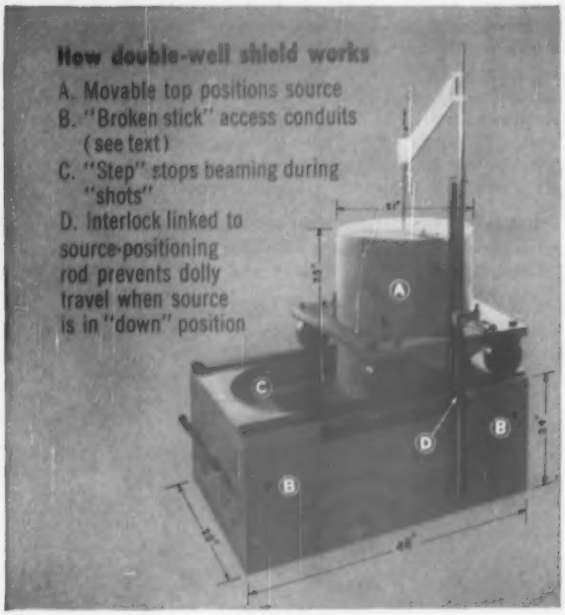
New Ray Stoppers

NATIONAL LEAD REPORTS RECENT DEVELOPMENTS IN LEAD SHIELDING

Double-test-well lead shield speeds sample irradiation by cobalt-60



To speed cobalt-60 irradiation experiments, National Carbon Company is now using a novel double-test-well shield built by National Lead*.

The device consists of two main parts.

The bottom shield, containing the wells, is stationary. During irradiation in one well the other is set up for the next experiment. The "step" prevents "beaming". "Broken stick" conduits provide access to the wells for themocouple leads or other test equipment.

The top shield, containing a new-type 2000-curie-source capsule, positions this source at the desired height in the appropriate test well. It also provides for a safety interlock. And doubles as the shipping container for the source.

Design meets four basic objectives

Both shields are carefully designed weldments . . . case, mild steel; cavity linings, stainless steel . . . filled by a special National Lead process with high-density, void-free lead. This construction and the geometry provide for:

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2. Reproducible or variable irradiation rates;
3. Uniform dosage over the surface of large samples;
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*From a design developed by J. A. Ghormley and C. J. Hochanadel at Oak Ridge National Laboratory, operated by Union Carbide Nuclear Company for the AEC.

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←This unit (one of a pair made by National Lead) is used as a storage cask for Co-60 in addition to its other functions.

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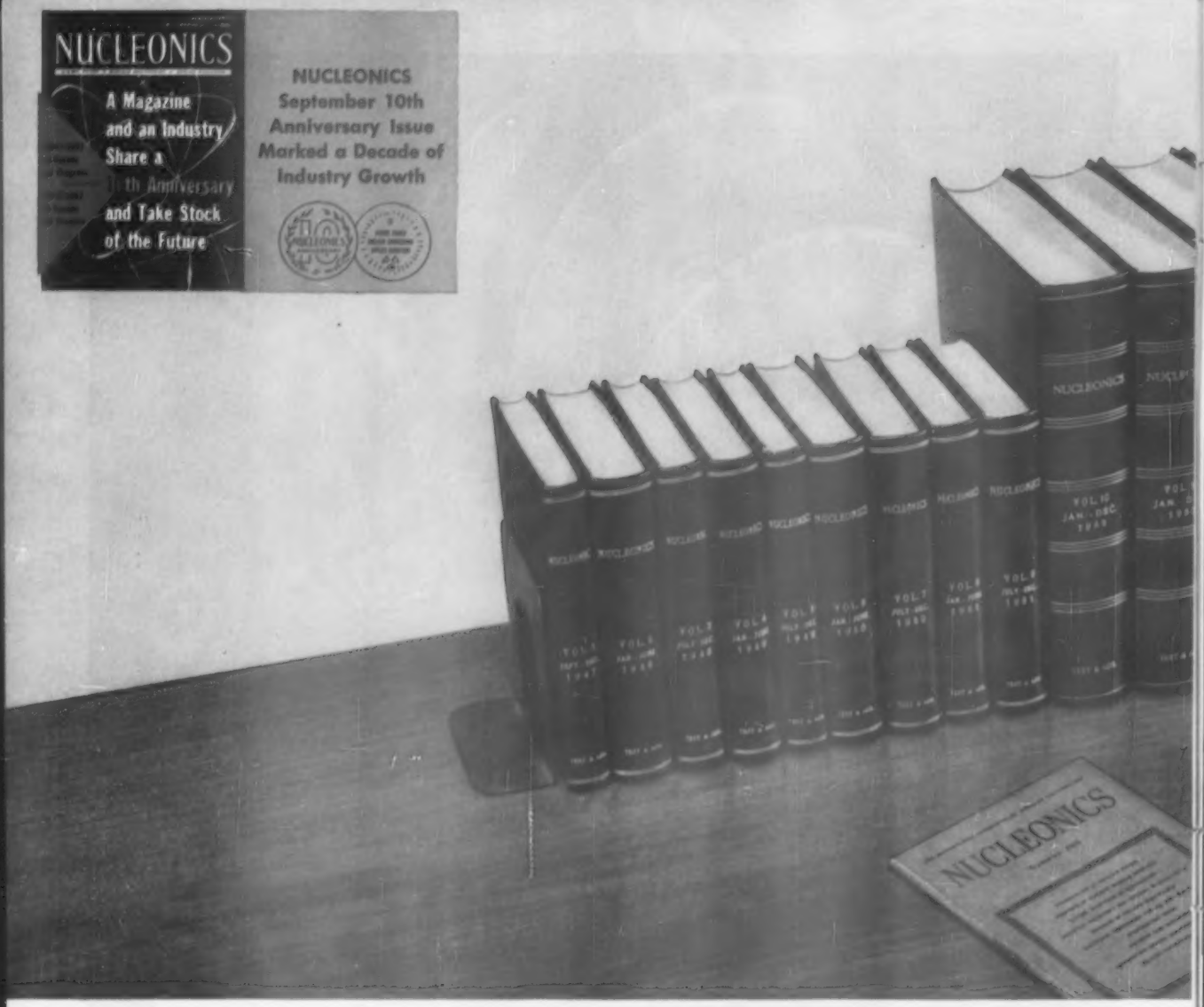
NUCLEONICS
A Magazine and an Industry Share a th Anniversary and Take Stock of the Future
NUCLEONICS September 10th Anniversary Issue Marked a Decade of Industry Growth
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They echo a challenge . . . one that makes us review the purpose of NUCLEONICS: to serve as the industry's only *authoritative* source for the newest developments in Atomic Power, Nuclear Engineering and Applied Radiation. This is the challenge of the *next* 10 years!

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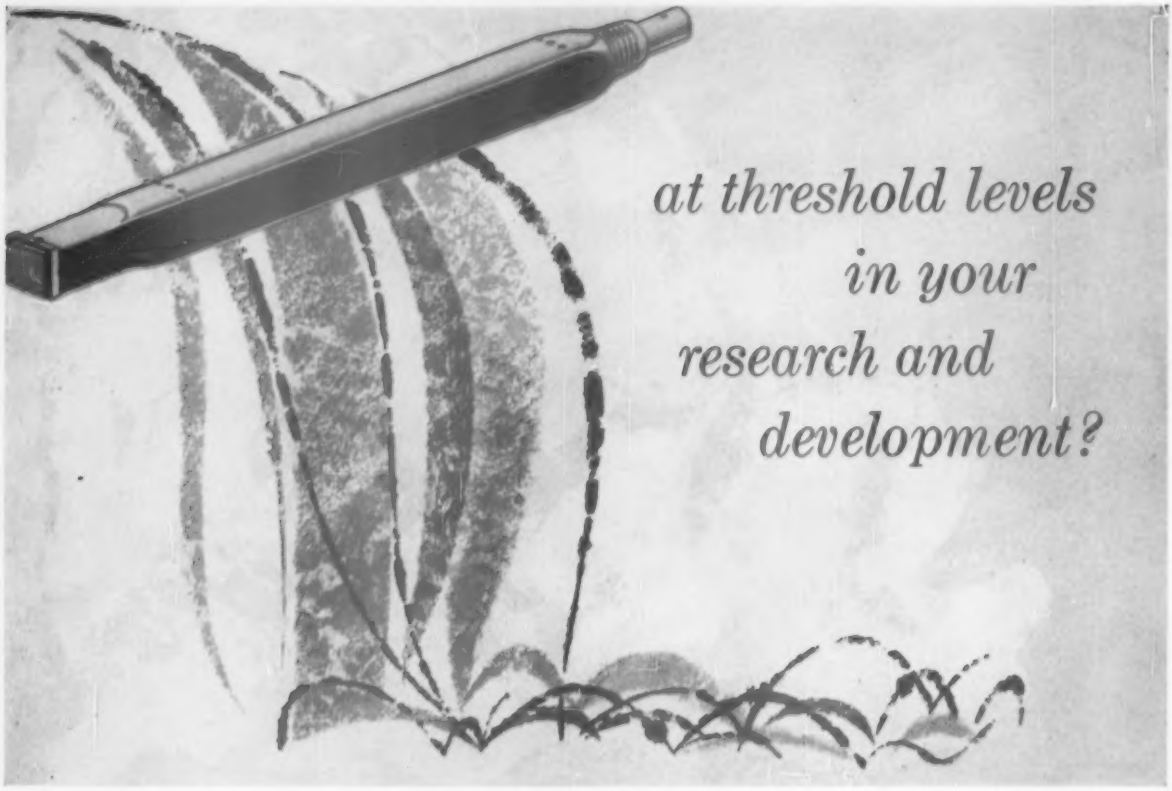

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ROUNDUP of Key Developments in Atomic Energy

Government, Industry Seek Solution on A-Power Problems

In 1949, Enrico Fermi predicted to the Joint Committee on Atomic Energy that: "[Competitive atomic power] is not around the corner . . . On the other hand, perhaps, one should not take necessarily a too pessimistic view about this situation either . . . I find it really extremely difficult to strike the right note between optimism and pessimism."

This month, at the time of the fifteenth anniversary of Fermi's Stagg Field reactor, the atomic energy industry is finding it almost as difficult as did Fermi to assess the United States program. Some observers view the present state of the industry (NU, Nov. '57, 21, 22) as rather bleak. Others see the atomic energy industry in a situation similar to that which the country as a whole finds itself as a result of the Soviet satellite developments—being forced to re-examine its objectives and the means for achieving those objectives.

Rather than being awakened by a *sputnik*, the atomic energy business is having its eyes opened by the economic realities of nuclear power. The industry is coming to the realization that, for the short-term picture, national policy considerations may well have to dictate the pace of the U. S. program rather than economic ones.

And so, all eyes are turned to Washington where the two bodies that hold the future of the industry —the Joint Committee and the Atomic Energy Commission—are desperately seeking to make some sense of a confused situation, with each body seeking its own answers. The subcommittee on research and development of JCAE, chaired by Rep. Melvin Price (D-Ill.), held informal, closed-door seminars with reactor designers and builders on Nov. 21 and 22. And AEC held meetings with the same groups plus utility representatives (not invited to the JCAE sessions) on Dec. 3, 4, and 5. Probably symptomatic of some of the problems of the industry is the mere fact that two separate inquiries were held.

Both discussions started from the same premise—that foreign policy considerations should be the overriding influence in determining the future course of the U. S. reactor program—and addressed themselves to these basic questions:

1. What specific policy objectives should the U. S. have?
2. What program should be established to achieve these objectives?
3. What should the time scale be?

These questions are being asked of the industry at a time when many companies are coming to the realization that some form of government support may be necessary to sustain the industry in an interim "uneconomic" period. Or, to put it in another way, some industry people say that if the government believes a larger program than now exists is necessary, then government money will be needed to carry this incremental effort.

This point of view then raises another set of questions confronting the policymakers:

1. If there is to be a subsidy, what form should it take—support of research and development, capital costs, fuel costs?
2. Who should be subsidized—the manufacturers, the utilities, or, in a sense, the government?
3. Who should determine how the

PWR goes critical

At 4:30 a.m. on Dec. 2—15th anniversary of man's first nuclear chain reaction—the U. S.' first full-scale nuclear central station went critical. The $110-million Pressurized Water Reactor at Shippingport became cold critical at 116° F with eight (of 32) control rods withdrawn 34.13 in. from the pre-assembled core; calculations for that rod-setting had been for criticality at 110° F. The core is now thought to be slightly (up to ½%) more critical than expected. Full power will probably be attained before the month is out.

Britain completes a Calder sale

Britain's first export order for a nuclear power station has at last been received. Italy is ordering a 200-Mw(e) Calder-type gas-cooled reactor plant, for a reported $56-million. The order goes to the Nuclear Power Plant Co., one of the four original British industrial consortia. Ordering the plant is AGIP Nucleare (NU, Sept. '57, 28), a state-controlled holding company for atomic development. The reactor, to be set up in north Italy, will be natural-uranium fueled and cooled by CO_2 in six primary loops. The station will be similar to, but smaller than the Bradwell station (300 Mw(e) from two reactors) being built in England by Nuclear Power Plant Co.; the fact the firm now offers 200 Mw from one reactor suggests an improved version. It's expected Italian firms will do much of the civil engineering work. A seven-year period of collaboration is called for in the contract between the two companies. Britain's Atomic Energy Authority will be responsible for extracting plutonium from the spent fuel, which is to be returned to Britain for reprocessing. From Rome, NUCLEONICS received word that ENI (parent company of AGIP Nucleare) feels no "sale" was involved, but rather that the accord signed Nov. 20 was merely preliminary to the negotiation of a contract for a reactor to be signed sometime before June 30, 1958.

money is spent—the "donor" or the recipient?

4. What are the implications of subsidy for the long pull?

Still more questions arise in connection with the interests of U. S. industry in exporting reactors. Apart from the very basic matter of the size and importance of the foreign market and the possibility of subsidy, industry needs answers to these questions:

1. What kind of guarantees on fuel element performance can be offered to foreign customers?

2. How can foreign nations be encouraged to deal with the third-party liability problem?

This then is the background for the study and soul-searching now going on. The points of view of essentially three groups—JCAE, AEC and industry—must be synthesized into a workable program. But there appears to be agreement on only one thing—that something more needs to be done than is being done.

What to Do: JCAE

Actually, there is also agreement on another point: that if any real solution is to be found to the problems at hand, it is vital that there be a harmonious environment in which to solve them. More specifically, JCAE and AEC need to work hand in hand, in an objective fashion.

An important, constructive step in this direction was the letter from JCAE chairman Durham to AEC chairman Strauss calling for "a series of informal sessions between representatives of AEC and JCAE to explore areas of possible agreement and methods of cooperation in defining objectives and means of implementation." The letter also repeated Durham's earlier call for a ten-year demonstration power reactor program "to take over where the five-year experiment reactor program leaves off." Strauss responded immediately by holding preliminary talks with Durham and other JCAE members, but what transpired was not disclosed.

The $400-million government construction program proposed last year in the Gore bill by the Democrats was along the right lines, Rep. Price feels, to get the U. S. atomic power program moving faster. And the Democrats again this year plan to offer a big program. Although Price did not say so, it is known that he, Senator Gore of Tennessee and Representative Holifield of California began working on such a bill immediately after they returned from their recent visit to Russia. Gore has described the bill as proposing a "big" program. Almost certainly it will call for Federal construction of power reactors again, but the size of the proposed program is not yet fixed.

Along with an increasing number of policy-makers both in government and industry, Price believes the emphasis in the U. S. power program should be aimed at the international field. Every time the Russians put up a big nuclear power plant, says Price, they will be scoring another psychological victory over the U. S.

On the other side of the fence within JCAE are the Republican members, who oppose a program that would extend the government's activities in the field of electric power generation and distribution, and who feel it would be wrong for the U. S. to enter a "kilowatt race." On the positive side, their views were cogently put in a farewell speech by former JCAE chairman Sterling Cole (R-N.Y.) who retired from Congress this month after 11 years' service on the Joint Committee on Atomic Energy. Cole offered a six-point plan for determining what needs be done—six things he said should be done by February, when the "Sec. 202 hearings" begin.

Unexpectedly, perhaps, for a Republican long identified with conservative private-enterprise economic views, Cole has espoused the principle of a subsidy for nuclear power development: "It would appear that the time has arrived when we must all recognize that federal subsidies are essential to a broad development program in reactor technology. The costs and risks are too great for private capital to do it alone. The field is too broad and other costs of government too demanding for the government to bear the entire development cost.

"Therefore, it seems to me that AEC should immediately recommend to the Congress a formula for federal subsidies to atomic development projects which would apply to all without discrimination, public and private sponsors."

Cole's five other points were:

1. AEC, in consultation with the State Dept., should prepare "a clear statement of the over-all and specific overseas objectives that a strengthened or expanded nuclear power development program by the U. S. should meet," including for example the relationship between Euratom's target goals and U. S. development goals, and the role American suppliers might be called on to fill in the Euratom program. Likewise the Administration "should establish in broad terms the monetary

Daystrom, Cook out; Sylcor drops plant

The shake-out in the nuclear industry that became evident last month (NU, Nov. '57, 21, 22) continued with two firms—Daystrom and Cook Electric—dropping out, and a third—Sylvania-Corning—canceling plans for a major nuclear facility. This was the $5-million Andover, Mass., project that was to have been a center for development and production of nuclear fuel elements and core components (NU, Dec. '56, R6; Jan. '57, R12). The decision to drop the proposed center was due to economic conditions affecting the entire nuclear industry, said Sylcor president Lee L. Davenport: of the field's three markets, government projects have been cut back; peacetime uses have not developed as predicted, and the overseas market "is rapidly being taken over by other countries with less cumbersome regulatory procedures." Sylcor will go on producing fuel elements at its expanded Hicksville, L. I., plant.

Cook Electric Co., whose Nucledyne Corp. subsidiary was barely a year old and had just completed elaborate radiation laboratories near Chicago; and Daystrom, active in the research and training reactor field, decided not to stay the course. Activities of Daystrom's nuclear division will be coordinated into the firm's central research laboratories. Stephen F. Malaker, who had been director of the division, is being retained by Daystrom as a consultant. He was also named Professor of Nuclear Engineering at Newark (N. J.) College of Engineering, and also is setting up a consulting firm.

support such foreign policy objectives should receive in the next five years."

2. AEC, in consultation with its technical advisory committees and the engineering and manufacturing industry, should prepare an appraisal of progress under its five-year program (the PWR–EBWR–SRE–HRT–EBR-2 program) and, based on it, "prepare a new and expanded government-sponsored program . . . [This] might include considerable emphasis on the development of small power reactors—not as prototypes of systems which may be economic in larger scale-up, but of reactors specifically designed to meet the needs of the underdeveloped areas. I believe considerable attention should be given to what I call the 'unsophisticated' reactor—one operating at temperatures and pressures that permit the use of more complex materials of construction and simpler fuel elements."

3. AEC, in consultation with its laboratories and the chemical and metallurgical industries, should consider ways to expand the present research and development efforts on fuel elements and on related fabrication and reprocessing systems.

4. AEC, in consultation with the manufacturing industries, the military services and potential foreign users, "should prepare a construction program to maximize the availability of high-flux test reactors.

5. JCAE should appoint a small, permanent, non-political, broadly-constituted advisory panel, not only to assist in reviewing the proposals of AEC, but also to prepare independently its own statement of objectives of our national atomic power program and the recommendations for achieving those objectives.

What to Do: AEC

In a recent speech, AEC chairman Lewis L. Strauss said he was studying proposals on how to expand the civilian power program: "I believe that we are now justified, on the basis of the valuable technology acquired from operation of the partnership program to date, to undertake the building of additional 'first generation' plants of commercial size, in addition to those already under construction or contracted for. The Commission is presently studying proposals for such an expanded program, and will be assisted in its study by the country's best authorities on the subject of nuclear power."

A spokesman for the Commission amplifies this by explaining it will not mean a totally new program with a new formula so much as new definitions on the assistance that AEC is ready to give industry, as part of the "partnership" program. New means of giving assistance are also likely to be under consideration, as for example post-construction research and development aid on fuel elements for a reactor after it is already in operation. Strauss himself recalled that he had said a year ago that AEC's program should be the subject of an annual review and that it should be modified as might be found necessary to insure continuing rapid progress. It is this review that AEC is now beginning in connection with the meetings Dec. 3–5.

But this is as far as AEC appears ready to go; it joins in denouncing a "kilowatt race" and federally-supported power reactor building programs. Also, Strauss said last month, "at the present stage of nuclear art, full-scale experiments . . . on advanced reactors are not warranted."

Still holding fast to and justifying his previous course of action to date, Strauss declared, "I find great personal satisfaction in today's event [the dedication of GE's Vallecitos reactor] for it seems to me an irrefutable answer to those who contend that American industry cannot be entrusted with major responsibility in the development of nuclear power, that industry cannot be relied upon to shoulder financial risks associated with the present stage of the nuclear art, and therefore that this new but fast-growing industry should be dominated by the government—and fostered entirely upon taxpayers' money."

The chairman appeared to give a top-level "no" answer to the suggestion of his reactor development chief, W. K. Davis, that water-moderated and -cooled reactors, being further advanced, be pushed harder for commercial advantage (NU, Nov. '57, 22). "As yet," Strauss declared, "there has been no Commission decision to stress the construction of full-scale nuclear power plants of any particular type, in preference to other promising concepts."

Finally, AEC also has its eye on the foreign market: "Any expanded assistance, in my opinion," Strauss said, "should embrace not only nuclear power plants to be built for operation in the U. S., but should also include research and development assistance to American manufacturers in connection with plants designed and built by them in foreign countries."

What to Do: Manufacturers

From a largely anti-subsidy view, industry—at least the manufacturers—have come around to the point where today the top executives of practically all the leading reactor builders are known to favor subsidy. There are some in industry who hold that large demonstra-

New group formed on gas-cooled LMFR

American Nuclear Power Associates was formed last month to study a liquid-metal-fueled, gas-cooled reactor type being pioneered by Raytheon Manufacturing Co. Others in the group include Burns & Roe, Clark Bros. Co. division of Dresser Operations Inc., Griscom-Russell Co., and Rockland Light & Power Co. of Nyack, N. Y. They will develop a high-temperature, high-performance nuclear plant on which Raytheon has done a previous study for AEC (NU, Feb. '57, R1). This is a reactor fueled by a uranium-bismuth solution and cooled by helium. The group's present prototype reference design has an output of 57 Mw(th), 16.5 Mw(e); critical mass is 5.6 kg fully-enriched uranium (total inventory is only 7.6 kg). Helium at 500 psi enters the core at 900°, leaves at 1,300° F. Reprocessing interval is 4.8 years at 80% load factor, with only a uranium pellet added from day to day in between; no control rods are used, as the reactor does not run on excess reactivity. Therefore simplicity and ease of operation are among the plant's advantages—a physicist is not needed to run it. Construction timetable will depend on the priority AEC gives the project, since the group cannot afford to develop it alone and will have to seek AEC research and development assistance.

tion power reactor projects, though financed preponderantly by federal funds, should nevertheless be controlled by industry. This is an extreme position; the search for a formula is on.

In 1953 when JCAE held hearings on changing the (McMahon) Atomic Energy Act of 1946, one of the few witnesses in favor of subsidy—foreshadowing the conclusion to which so many of his colleagues have since been driven—was Chauncey Starr of Atomics International. Said he then (NU, Sept. '53, 14; 23)—and his views remain unchanged: "As the real market for central station-size atomic power plants is several years away, and their development is a national asset, government support of a technical program leading toward their economic performance would be justified. Such support could be gradually removed as a real industry develops . . . I think if you want to put this field on a basis that is not government-supported, what is going to have to be demonstrated is the competitive privileges of power from nuclear plants. Until such demonstration takes place, I believe government support of one sort or another is going to be required. If the government wants full-scale plants built, it is going to have to support this.

"My own feeling is simply that the government should pay for the manufacture of knowledge, but should not subsidize the manufacture of power. We feel that government support in the manufacture of knowledge is a justified thing."

A new and potentially very significant move in industry circles is the attempt—now being made by a leading manufacturer—to organize a pool of organizations that would jointly guarantee fuel-element performance. American reactor-builders' inability to issue such guarantees to date has been held by many observers to be a major stumbling block in the way of U. S. power reactor sales abroad. Although the U. K. Atomic Energy Authority steadfastly declines to confirm it, it has been frequently reported from prospective buyers of British reactors that the U. K. is ready to guarantee performance of its fuel elements.

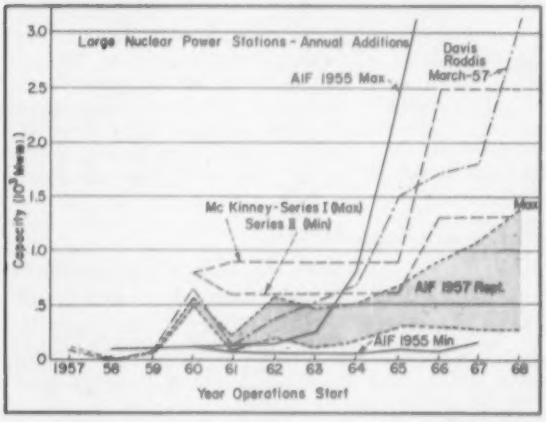
In this connection, another new straw in the wind is the report that the Export-Import Bank in Washington, which had earlier said it is ready to grant loans to overseas purchasers of U. S. reactors (NU, Nov. '56, R8), is now considering advancing funds to cover fuel elements and fuel-cycle costs.

What to Do: Utilities

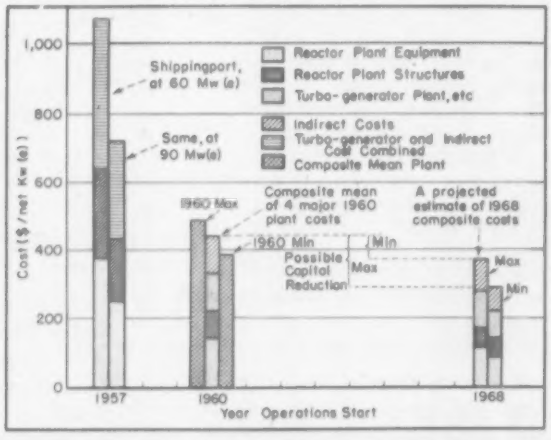
But while the manufacturers in increasing number now appear ready to regard with equanimity, if not as an outright necessity, the entry of federal subsidy into the reactor de-

Forum Reworks Its 1955 Industry Growth Survey:

1. Various Estimates of Future Construction



2. Capital Costs of Large Power Plants



The oft-quoted 1955 report by the Atomic Industrial Forum, "A Growth Survey of the Atomic Industry, 1955–1965," has been revised by the Forum, the revision to be published next month. A preview of the new findings was presented to the Forum annual meeting in October by the authors, Frederick H. Warren and William W. Lowe of Pickard-Warren-Lowe Associates. **Chart 1** above, comparing a number of estimates of annual additions to large nuclear power stations, shows that while the minimum prediction is somewhat higher than that of two years ago, the maximum is less optimistic by quite a wide margin for the years after 1964. However a higher early volume of large reactor plant construction than was in prospect three years ago is now seen.

Chart 2 compares best available data on capital costs for PWR and for Dresden, Indian Point, Yankee and PRDC's Fermi fast breeder shown in composite. Average total investment for these four, with indirect costs and interest during construction included, is $430/kw(e); estimated maximum and minimum range of capital costs for an average large plant of 1968 was estimated at $360 and $280 respectively, by projecting possible cost reductions (omitting any dollar-inflation effects). The reductions result from increased unit capacity and reduced cost for components. While costs are now rising and no turning point is yet marked, the authors justify predicting cost reductions on the ground the current cost increases are "no longer a great mystery," having been caused by additional development costs needed to solve technical problems unexpected when the preliminary estimates were made; additional engineering needed to get satisfactory system designs; increases in engineering, pro-

velopment picture, the investor-owned utilities as a group are far from such a position. Traditionally more conservative in their economic outlook than the manufacturers (perhaps because of a geographically more localized horizon?) they still see any increase in federal support as a major threat to their very existence in the frame of reference of the private-vs-public power struggle. In particular, they look upon the preference clause in the 1954 Atomic Energy Act (Sec. 44) as invidious, and take the position that if subsidies are necessary and if the prohibition on subsidies in the Act is to be altered, then the preference clause also should be deleted and the availability of government assistance to both private and publicly-owned power agencies without discrimination be guaranteed.

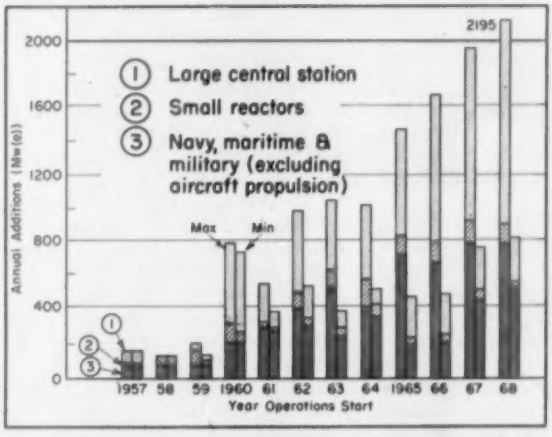
What to Do: Reactor Experts

According to some of the top reactor designers, engineers and physicists from industry and the national laboratories who advise the government, the main problem—contrary to the prevailing view for which AEC has obtained credence—is lack of money.

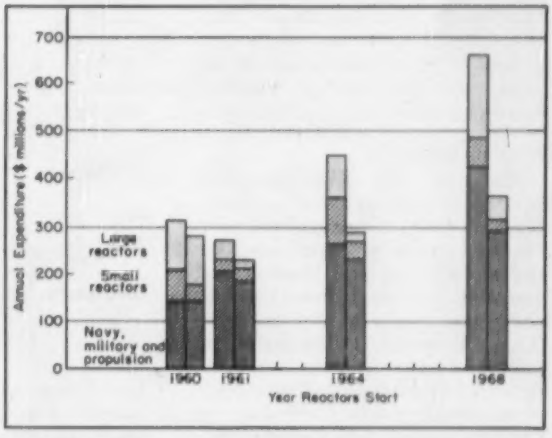
This small experts' group feels, NUCLEONICS learned, that AEC research and development spending is lamentably inadequate, while—again contrary to general belief—manpower is not a major problem. If the only aim of our reactor development program is economic power, they say, then we are doing enough; but if foreign policy considerations are included in our reactor goals, then Congress and the Administration must loosen up on money for research facilities and research programs. Ever since the war, say the experts, Congress has been reluctant to give money for supporting facilities and laboratories; we are woefully behind on test reactors; we need to put more money into engineering research, into fuel technology. For example, the engineering branch of AEC's reactor development division now gets $8–10-million a year; it needs $50-million, the experts hold. Again: one of our most urgent needs is to develop a good, high-performance fuel element for water reactors, and a concentration of research effort here is sadly lacking. We need to build more full-scale power reactors; if we are to have economic nuclear power, here or elsewhere, two or three generations of big reactors may be required (beyond the stage that Yankee, for instance, represents as the "daughter" of the Shippingport PWR). Industry cannot pay for such a load of sub-economic plants, say the experts. Finally, there is a need for long-term commitments for funds, to give the assurance to industry that Congressional year-by-year appropriations cannot give.

Maximum Predictions Are Now Much More Modest

3. Reactor Power Projections—Kilowatts



4. Reactor Power Projections—Dollars



duction and inspection costs of components due to unique nuclear specifications; and 5–10% average general escalation per year in materials and equipment costs.

Chart 3 compares total U. S. market projections for large central station reactors; small power and miscellaneous types; and naval, maritime and military reactors exclusive of aircraft propulsion units. The second, small-reactor group includes power reactor prototypes; general test reactors such as MTR and ETR; radiation application and process heat units; and U. S. built small reactors for export. The over-all size of the naval program was taken from the defense budget, with seven submarines (fiscal 1957) taken as a maximum and four submarines (1958) as a minimum; the estimates are consistent with Adm. Rickover's statement that the Navy would "have by 1963 1.5-million kw of power in operation, and an additional million in ships under construction." The Navy program "represents today the most assured market for reactor construction in this country," says Warren. These power projections can be converted to dollar projections by multiplying by unit dollar values for each reactor group: for large stations, the values shown in Chart 2; for naval reactors, a range of $700–550/kw(e) was estimated; for small reactor systems, $600–500/kw(e).

Chart 4 makes these conversions. The values cover only the nuclear system and its related structures, but no costs for power generation, fuel cores, nor indirect costs and development expenses. The naval and propulsion reactor group is now seen to comprise between 40% and 80% of the total estimated gross business. Aircraft propulsion is excluded as being too speculative to fit the other estimates.

Sweden Pushing Power, Space-Heat Reactors; Two Under Construction, Third Being Designed

STOCKHOLM

One of the most purposeful programs for development of civilian reactors outside the Atomic Big Five (U. S., Britain, USSR, France and Canada) is that being pushed in Sweden. In fact it may be said that this country, together with Belgium and neighboring Norway, is furthest advanced among the smaller powers in the reactor art.

Sweden's atomic program envisages the building of a full-scale nuclear power station and 5–6 plants for district heating over the next 10 years, with some of the latter also used for power output through back-pressure operation. These plans were recently confirmed by Harry Brynielsson, president of Atomenergi, Sweden's atomic energy company founded in 1947. Atomenergi, which is ½ state-owned, ½ privately owned, has asked governmental appropriations of 138-million kronor ($26.6-million) for the 1958–59 fiscal year—double the funds granted in the current budget.

Atomenergi is not likely to get all the funds it seeks in this state economy where reduction of expenses and of public spending are watchwords. But the company can look forward to a considerable expansion of its activities. Private industry also is increasingly entering the atomic field.

Program Bases

Since it was first launched, Sweden's atomic program has been based on the use of natural uranium —which can be produced locally— and of heavy water. None of the experience amassed thus far has given any reason to change these plans. However the economy of natural-uranium reactors could be improved by adding small quantities of imported enriched uranium or locally-produced plutonium. Reactors for ship propulsion and for small space-heating plants are likely to use enriched uranium or plutonium, and light water moderation.

Raw materials for the country's nuclear program are available domestically from the low-content uranium shale deposits in Central Sweden. Exhaustive prospecting is being carried out elsewhere in the country by Atomenergi, the Swedish Geological Survey and private companies, and these may result in new finds of importance.

In 1953 Atomenergi placed in operation a uranium extraction pilot plant at Kvarntorp with a capacity of 5 tons/yr. This capacity has now been doubled, and based on experience gained with it a new extraction plant has been projected, with a capacity of 120 tons/yr of uranium. This is estimated to cover the country's needs until about 1965. About 400,000 tons of shale will have to be mined annually to attain this output, which shows the low uranium content of Sweden's shale deposits. However, reserves have been estimated at 1 million tons of uranium (NU, Jan. '56, 15).

Research Reactors

Sweden's first reactor, R-1 (formerly also called *Sleep*, for Swedish

Low Energy Experimental Pile) was completed in July 1954 by Atomenergi at an underground location in Stockholm. It was originally designed for 300 kw(th) but is now continuously operating at 600 kw, and a further step-up is seen possible. R-1 is natural-uranium fueled and heavy-water moderated and cooled, the D_2O being circulated out of the core and through an air blast heat exchanger. It has been used for basic physics research, as for example on scattering of very cold (~2° K) neutrons (this last work was performed by a group headed by Harry Palevsky of Brookhaven National Laboratory, who just completed a nine-months stay in Sweden). R-1 is also being used increasingly by outside industries.

Work is proceeding on Sweden's second reactor, R-2, which when completed will probably be Europe's most powerful research reactor. This is a 30-Mw(th) ORR-type (tank-in-a-pool type) materials-testing unit that ACF Industries is building for Atomenergi. It will have 10^{14} maximum thermal flux; will be heavy-water moderated and cooled, and will use enriched uranium as fuel—the Swedes are hoping to get their U. S. bilateral amended to enable them to obtain 90% enriched uranium for R-2.

The reactor will include pneumatic and hydraulic rabbits, a graphite thermal column, a D_2O jacket around the core to increase the volume having a useful neutron flux. The lattice will permit arranging five different loops through the core, several (but not all) of which can operate simultaneously. R-2 will be used as a tool for investigating fuels and other reactor materials, with work aimed primarily at improving fuel economy; also for basic research in neutron physics continuing the type of work that has been done with R-1. A fast chopper and a crystal spectrometer are among the ancillary items of equipment that will be installed.

The building that will house R-2 is already under construction at Studsvik on the Baltic coast 70 mi S of Stockholm, and ACF is to start shipping components next year. Under its very tight construction schedule R-2 is slated to go critical toward the end of 1958. The pool lining, heavy water ducts, and valves are to be made in Sweden; the instrumentation and control console have been subcontracted to the Dutch firm Phillips N.V. and its Swedish subsidiary. The control-rod drives, however, will be furnished by ACF. Atomenergi, which will operate the reactor, has assigned a man to ACF's shops, and the American firm has a man stationed in Sweden.

Under consideration is a plan to install a critical facility in one of R-2's other pools, just for studies on R-2 fuel elements. Aside from this, however, there will definitely be one zero-power reactor at Studsvik, designated R-0. Its function will be to study different properties and different arrangements of natural uranium fuel elements and element clusters, at different temperatures of D_2O moderator, and with

water or gas as coolant. It will operate at first without a reflector; later a reflector will be added.

Building costs for the Studsvik installation are estimated at about 25-million kronor ($4.83-million), one-third of which will go for the ACF unit.

Heat & Power Reactors

In more or less advanced stages of planning are two space-heat reactors and one for electric power generation. This represents something of an abridgement of earlier plans (NU, Dec. '55, 15; Jan. '56, 15), to the extent that original plans for two reactors designated R-4 and Eve have been telescoped into one project.

R-3, also known as the Agesta reactor, will be a space-heat or dual-purpose reactor serving Farsta, a brand-new, 12,000-apartment housing development on the south side of Stockholm. Conceived originally as a straight space-heat unit, R-3 was successively converted to a heat-and-electricity design and back to heat-only; today it is still under discussion whether or not R-3 is to supply 15 Mw of electricity in addition to 70 Mw of heat.* The design is to be frozen early in 1958; in the meantime, however, rock blasting last month marked the beginning of construction work at the site. R-3 will be fueled with natural-uranium oxide elements (core loading will be about 11 tons of oxide) and moderated by heavy water. It will probably be cooled by means of a re-entrant tube system (pressurized tubes) surrounding the fuel elements, carrying heavy water in an independent loop. An undertaking of Atomenergi in cooperation with the Stockholm municipal electricity board, R-3 is scheduled to be completed in 1960.

Adam, one of two reactors planned jointly by Atomenergi and Sweden's State Power Board, is a 75-Mw(th) low-temperature heat reactor now under construction at Västerås, near the electrical-equipment factories of Sweden's huge ASEA Co. which has the design contract and will probably also build Adam. Scheduled also for 1960 completion, Adam—like R-3—will be natural uranium oxide fueled and heavy water moderated, but unlike R-3, Adam's core will be housed in a pressure tank and cooled by circulating heavy water in guide tubes around the fuel elements.

Eve was the State Power Board's project for a 100-Mw(e) power plant. R-4 was Atomenergi's plan for a 75-Mw(e) power unit. When the two organizations decided to pool their efforts on Sweden's first large nuclear stations, they were able to avoid eliminating either plan entirely—and thus keeping both parties happy—by the simple expedient of designating the station Eve and the reactor R-4. Site has not yet been decided, but it is likely to be underground in the mountains of south-central Sweden. Output will be 100 Mw(e), and cost is estimated at 180-million kronor ($34.7-million). Fuel will be natural uranium; moderator, D_2O; coolant has not yet been chosen but both liquid and gas cooling are under consideration. The plant was to have been completed by 1963, but 1964 is now considered more realistic.

Atomenergi's requests for funds are now under consideration by the government, and a final decision by the Minister of Finance is due before Christmas. Construction schedules can then be worked out in detail.

Further plans include a reactor for the pulp industry.

Industry Collaboration

Spokesmen for Atomenergi emphasize that accomplishment of these and further goals in Sweden's atomic energy program will require active cooperation on the part of domestic private industry. The facilities described above are financed largely with state funds; they are seen as prototypes that must be developed further by industry, and scaled-up versions built by industry, for municipal authorities, power companies, and steam-consuming manufacturing plants.

Leading private companies in the atomic field in Sweden are ASEA (whose initials stand for the Swedish words for "General Electric Company," but which has no connection with the U. S. GE), the highly-diversified Johnson Concern (which is cooperating with Westinghouse), and Nydqvist & Holm. All have established atomic power divisions that will be resorted to more and more for designing and building components for new Swedish reactors. ASEA, in collaboration with Atomenergi, is experimenting with the production of heavy water. ASEA is also reported to be investigating the possibilities of developing a "medium temperature" reactor. Addition of an oil-fired superheater could make such a plant suitable for both heat and power-producing purposes. Finally, ASEA's plans are also said to include design of a small heat plant of 35–40-Mw(th) using enriched uranium.

Sweden is also doing considerable work on design of reactors for merchant ship propulsion (NU, Nov. '57, 87).

Fusion

Kai Siegbahn of the University of Uppsala has, with ASEA support, performed some work on gas discharges using very large capacitors and in this way has repeated results reported earlier by the Russians. At the Stockholm Institute of Technology, H. Alfven (a long-time expert in magnetohydrodynamics) and L. Hulthen have been carrying out experimental and theoretical investigations of more fundamental character in the thermonuclear field. However, spokesmen have said Sweden will wait until the 1958 Geneva conference fusion sessions before firming up her own thermonuclear program.

Weapons?

As has been reported abroad, the chief of the Swedish General Staff has on two separate occasions asked that nuclear weapons be incorporated in the Swedish defense arsenal. Up to this time the Swedish government has taken no decision whatsoever on this question, contenting itself with the statement that the effects of nuclear weapons must be studied in order to make Swedish civil defense as well prepared as possible for the contingency of atomic warfare. It is known, however, that the government considers it essential to foster the peaceful uses of atomic energy to the maximum possible, and may be postponing the decision on A-weapons. Some Swedish observers believe this is all the more plausible since it may be felt that development of a number of large reactors, with their attendant plutonium production, could make an A-weapon program in the future easier, so that no time would have been lost.

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* Originally, the R3 design was conceived in two stages: an early stage, R3a, was to produce 90 Mw heat; later, the reactor was to be altered by replacing the initial pressure tubes with newer ones able to withstand considerably higher pressure. In this second stage, R3b, the reactor would be able to produce not only 75 Mw heat but also 15 Mw electricity by means of a back-pressure turbine.

Britain to Concentrate On Fast Reactors

After 1980 Britain will build only fast reactors for power, John Gaunt, British atomic attaché in the U. S., said last month. The present Calder-type reactors will be built until 1965; reactors completed 1965–70 will use beryllium-clad fuel elements; 1970–75 beryllium-canned oxide fuel will take over; by 1980, half of all power reactors will be fast reactors. From then on, thermal reactors will start being phased out. The reason for this, he said, is that the best burnup to be hoped for with uranium-burning thermal reactors is 3%, and at present fuel costs are ⅓ of total power cost; therefore any real improvement in fuel cost for uranium reactors must be done with fast units.

The sphere for the Dounreay fast reactor, Britain's first, is completed, and the first core is being fabricated; the secondary circuit is almost finished. The reactor will go critical next summer. It will be used first for zero-energy experiments, then for oscillator experiments at all power stages, and when it has operated at design power it will be tried with plutonium elements. However Dounreay is strictly an experimental power reactor; the first prototype fast reactor will not come until the late 1960's, Gaunt said.

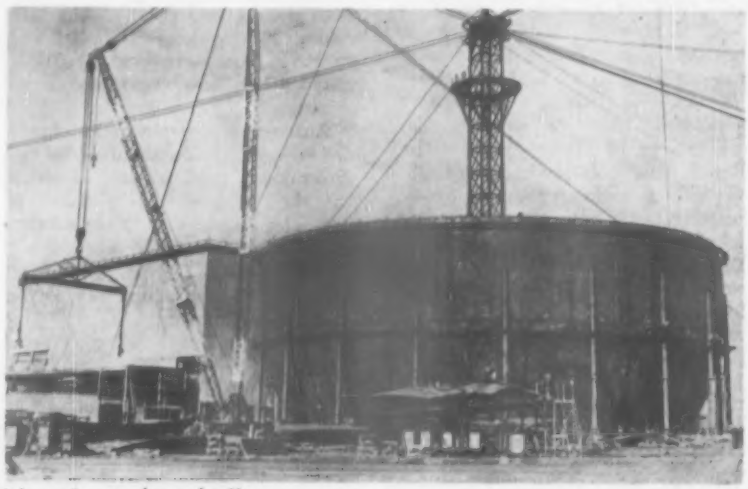
Camden Yard to Build 'Savannah,' First A-Ship

New York Shipbuilding Corp. was awarded a $20,908,774 fixed-price contract to build the first U. S. nuclear merchant ship. The vessel (NU, Nov. '57, 78) will be called N.S. *Savannah*, according to a White House announcement; the new designation N.S. (for nuclear ship) replaces the familiar S.S. (steamship). The Camden, N. J., yard will lay the keel for *Savannah* early next year, for 1959 launching and delivery by mid-1960. The contract covers construction of the ship, and installation and testing of her Babcock & Wilcox reactor propulsion system. Among the ship's features will be provision to permit tours to view the ship's machinery in operation, and a single console in a central, air-conditioned control room for remote-control operation of the entire power plant, including both nuclear and conventional machinery.

Two Power Reactors Rising Fast

Putting the lid on

The dome or "beanie" of the gastight containment building for the Enrico Fermi fast breeder at Lagoona Beach, Mich., being lowered into place. Pressure vessel is 72 ft dia, 120 ft high (about 50 of it below grade). Building has been pressure-tested at 40 psi. View is looking south; compare site photo NU, Sept. '56, R4



It's going to be a ball

The 190-ft dia sphere to house the Dresden Station boiling water reactor is already one-third complete. Sphere, of special steel plates from 1.25 to 1.4 in. thick, will rise 150 ft above plain 47 mi SW of Chicago; 40 ft will be below grade. In left background, turbine building to house 180-Mw turbo-generator is almost completely enclosed. Progress since start of major construction six months ago has been swift (compare photo NU, June '57, 22; also sketch of completed plant, Nov. '56, R8)

1958 Geneva Parley Planning Under Way

Sigvard Eklund, technical director of Sweden's Atomic Energy Co., has been chosen by the U. N. as secretary-general of the 1958 Geneva Conference—the second International Conference on the Peaceful Uses of Atomic Energy. This is the post held in 1955 by Walter G. Whitman. AEC has set up an Office for International Conference under Edward R. Gardner to plan and direct U. S. participation in the conference; L. D. Percival King of Los Alamos Scientific Laboratory has been appointed Technical Director (the post held last time by George Weil). Invitations have gone out through 4,000 universities, companies, etc., for technical papers.

Reactor News

CANADA'S NRU GOES CRITICAL

After six years of design, engineering development and construction, Canada's high-flux NRU research reactor went into operation at Chalk River, Ont., Nov. 3. It will continue to be operated at low power for several weeks of test experiments, after which power will be increased to the rated capacity of 200 Mw(th), five times the power of Canada's workhorse research reactor, the neighboring 10-year old NRX. A triple-purpose reactor, NRU will not only provide research, testing and experimental facilities for nuclear power development—it will also produce both plutonium and radioisotopes. According to Atomic Energy of Canada Ltd., its builder, NRU will be unsurpassed as a research and testing tool: "no other known reactor has such a high density of neutrons combined with such a large capacity for experiments." It will accommodate any fuel element assembly now being considered for nuclear power plants. The U. S. AEC and U. K. AEA, both of which have benefited from NRX, have both already asked for testing space in NRU. More than 100 Canadian firms have contributed to making components for the $57-million, heavy-water-moderated and -cooled, natural-uranium-fueled reactor. Unlike NRX, NRU can be refueled without shutdown.

YANKEE GETS CONSTRUCTION PERMIT

After two months in the toils of legal technicalities (NU, Sept. '57, 30; Oct. 25), Yankee Atomic Electric Co. finally obtained AEC construction permit No. 5 on Nov. 4. A concurrent memorandum of opinion signed by all five commissioners said "the construction permit must be provisional at this stage of the proceeding principally because Yankee has not completed the design of the reactor and certain features of proposed facility present unresolved questions which might affect its safe operation. . . . These questions include: the addition of neutron absorbers; intentional design into the reactor of nucleate boiling; and large plutonium build-up. . . . We agree [with AEC hazards evaluation chief Clifford K. Beck and the Advisory Committee on Reactor Safeguards] that the experimental program outlined in the application will probably produce the information needed."

The memorandum noted that Yankee estimates its construction cost at about $57-million (exclusive of $5-million AEC research aid); this compares with $35-million originally estimated.

CATALYTIC GETS PUERTO RICO CENTER

Catalytic Construction Co. has been named architect-engineer for the Puerto Rico Nuclear Center (NU, Nov. '57, 28), including the reactor building to house the pool-type research and training unit. The firm to design, fabricate and install the reactor itself has not yet been selected.

CAROLINAS-VIRGINIA GROUP PICKS SITE

Parr Shoals, S. C., on the Broad River, about 25 mi N of Columbia, has been chosen as the site for Carolinas-Virginia Nuclear Power Associates' 17-Mw(e) heavy water power reactor. CVNPA's proposal is now before AEC for approval, and under the amendment to the law passed this summer, Congress will have to approve specifically all AEC aid to the group.

GE TESTS SPANISH POOL REACTOR

The 3-Mw(th) pool reactor GE is furnishing the Spanish nuclear research center at Moncloa near Madrid has been successfully tested at GE's Vallecitos Atomic Laboratory. It will now be shipped and assembled early next year; criticality at Moncloa is set for March.

MTR TO GET PLUTONIUM CORE

After use of several plutonium fuel elements in the Materials Testing Reactor core (NU, Aug. '57, 26), an entire core of plutonium will be tried, Phillips Petroleum Co., MTR's operators, have told NUCLEONICS. Although "there are no plans to convert MTR to the use of plutonium as a fuel on a permanent basis, present plans do contemplate, however, the loading of one plutonium core into the reactor to gain whatever experience may result thereby. Several modifications in the control system will be required to effect the change-over. We are presently in the initial discussion stage as regards the fabrication of the plutonium fuel elements and control rods. It is not known as yet who will fabricate the fuel elements, although Oak Ridge National Laboratory seems to be the most likely possibility. It will be several months at least before the Pu core loading can be made. We presently see no advantages in using plutonium in MTR; actually, there will be several disadvantages so far as this particular application is concerned. However, we think it will be possible to operate the reactor reasonably safely with a plutonium core and can thereby [be] the first facility to accomplish this historic objective."

ACF IN TRAINING REACTOR MARKET

A one-watt "pre-fabricated" pool-type training reactor has been placed on the market by ACF Industries at $135,000–150,000, including shielding and instrumentation. Exterior measurements are 10 ft high, 14 ft 7 in. deep, 16 ft 5 in. wide, including concrete-block shield.

TWO REACTOR DEDICATIONS ON COAST

AEC's 6.5-Mw(e) Sodium Reactor Experiment, built by Atomics International at Santa Susana, near Los Angeles, was dedicated Nov. 14. On Nov. 25, General Electric's 5-Mw(e) developmental boiling water reactor at Vallecitos, near San Francisco, received its formal baptism. Both are forerunners of larger plants of the same type—the former of the Nebraska Consumers Public Power District 75-Mw(e) reactor, the latter of Chicago's 180-Mw(e) Dresden station.

AIR FORCE CANCELS P & W CONTRACT

Pratt & Whitney's Air Force contract for development of the J-91 nuclear plane engine was canceled four months ago, it was disclosed by the Air Force in a letter to Connecticut Senator Purtell.

BRITISH DISMANTLE ZEUS

Zeus, the research reactor used for design studies of the Dounreay fast breeder, has been dismantled. Its shield is being adapted for neutron studies with another undisclosed reactor.

BURNS & ROE ON DUAL-PURPOSE JOB

AEC has asked Burns & Roe to make the preliminary design for the plant and supporting facilities of the new Hanford plutonium reactor (NU, Nov. '57, 25); GE is designing the reactor itself.

ROUNDUP

World News

West Germany Plans 4–5 Power Reactors

The minimum program for German nuclear development, submitted to Atomic Minister Siegfried Balke by the Atomics Commission, calls for construction of 4–5 power reactors with a total capacity of 500 Mw(e), costing about $240-million, and to be completed in five years. The reactors are to be built in close cooperation with foreign firms, and are to include different types so as to gain maximum experience. Balke told a press conference he feels chances are dim for rapid utilization of nuclear energy in Germany if industry does not show greater interest. Later, the director of the Hamburg Public Power Supply Co. said northern Germany can expect to have a 100-Mw(e) nuclear plant "probably within the next ten years."

Winfrith Heath Construction Started

Construction began Nov. 25 on the new $56-million reactor research station at Winfrith Heath (NU, March '57, R10). It will function essentially as an out-station of Harwell, which will tend to concentrate more and more on purely fundamental research, while Winfrith Heath will work almost entirely on design and development of new types of reactors. British General Electric Co. already has a contract for one of the first reactors to be built at Winfrith Heath: a zero-energy, high-temperature gas-cooled breeder experiment burning a mixture of thorium and U^{233} (NU, Sept. '57, 31). It will have hollow unclad ceramic fuel elements, probably uranium oxide, and will cost $5.6-million. Another advantage forecast for this design is that it will require only 1/100th as much graphite moderator as the Calder-type reactors. Altogether some eight experimental reactors may be built at Winfrith Heath in the next 10 years, including two ship-propulsion prototypes and an aqueous homogeneous.

Japanese Utilities Organize

In a move expected for many months, Japan's nine electric power firms have set up the Japan Atomic Power Generation Co. to import and operate a nuclear station. The utilities are investing 42% of the new firm's paid-up capital of 1-billion yen (eventually to be 10-billion); 20% each comes from the semi-governmental Japan Electric Power Development Co. and a group of manufacturers interested in nuclear energy; and the remaining 18% from promoters of the new firm and the general public. Study missions are to be sent to the U. S. and Britain by the new firm.

Two International Trade Pacts in France

France's first important international trade agreement in the nuclear field (NU, Nov. '57, 17) was announced last month: the big French electronics firm Thomson-Houston, and an associated firm, the turbine builder Alsthom Co.—both of whom do contract work for the French AEC—signed an agreement with General Electric providing for exchange of technical information and technical aid on nuclear matters. Thomson-Houston was once a GE subsidiary, and still has close ties in the electronic and radio fields. Another agreement expected to be announced shortly is between Westinghouse and the huge Schneider-Creusot iron, steel and heavy machinery firm.

French to Start A-Ship

The French AEC has announced plans for construction of a prototype nuclear-powered tanker of 40,000 tons, with propulsive power of about 20,000 shp. Pressurized water, boiling water and gas-cooled reactors are under consideration; a decision is due in March.

Icebreaker Reactor Checked Out

Tass reported last month that the 40,000-hp nuclear power plant for the Soviet icebreaker *Lenin* has been completed and successfully tested. It did not say whether or not it has yet been installed in the hull, on the stocks in a Leningrad yard (NU, Aug. '57, 18).

Greek-Yugoslav Atomic Cooperation

Yugoslav nuclear scientists visiting Greece have offered to train 20 Greek nuclear physicists and technicians in Yugoslavia's atomic centers. The Greek AEC has offered reciprocal facilities and invited Yugoslavia to send scientists to study at the first Greek research reactor now being built near Athens, and to attend a series of lectures on isotopes. The two countries have also laid the groundwork for collaboration on exploration, processing and technology of radioactive ores. Greece's AEC expects to benefit considerably from this cooperation, as Yugoslavia already has several large nuclear research centers and 2,000 scientists and technicians in the field.

Czechoslovakia Plans Big Power Plant

Czechoslovakia last month unveiled a model of its first nuclear power plant, a 150-Mw(e) unit to be built at a site in Slovakia. It will be designed and major components furnished by the Soviet Union, which has in fact licensed to the Czechs an undisclosed type of power reactor that may be built later in the USSR: this has given rise to a belief that the Russians may be using Czechoslovakia as a testing ground for a new reactor. The announcement came just after Czechoslovakia started up her first research reactor (NU, Nov. '57, 26), and was accompanied by a statement that henceforth the total annual increase rate in Czech electric power requirements is to be met exclusively from nuclear power plants.

Germans Endow Thorium Research

The Bonn Atomics Ministry has supplied DM 370,000 ($88,000) to the inorganic chemistry department of Munich University for a thorium research program. The money will be spent for developing processes for extracting thorium from ore and for U-Th separation methods. It is held thorium is more advantageous than uranium for German reactors as U^{233}-breeding would save the cost of U^{235} enrichment.

Finland Buys Reactor Materials

The YDIN Nuclear Power Association of Helsinki has concluded negotiations to purchase materials for an exponetial pile, Finland's first (NU, Oct. '57, 28). Britain's Atomic Energy Authority has an order worth $72,800 covering most of the materials needed; the graphite may be bought in the U. S.; and $6,000 worth of instrumentation was ordered in the U. S., Sweden and Holland. The state-owned woodworking firm Enso-Gutzeit is constructing the building that will house the pile, as the state's contribution to a private project.

News in Brief

Northern States Signs Contract

AEC has signed the first third-round power demonstration program contract, with Northern States Power Co. of Minneapolis. Total cost of the project for a 66-Mw(e) Allis-Chalmers advanced boiling water reactor (NU, March '57, R8; Sept., 30) is put at $28-million, including research and development. AEC would contribute $6-million for research and development, and waive fuel charges. No site has yet been named, but Northern States says it is thinking of one on the Big Sioux River, NE of its present plant at Sioux Falls, S. D. Northern States and 10 neighboring utilities have formed Central Utilities Atomic Power Associates Inc. to participate in the project; the other 10 will contribute $3.6-million to the over-all cost, and will get technical data and training opportunities. The firm plans to complete the reactor by June 1962, and will operate it at least five years, under the AEC contract.

AEC Phasing Out St. Louis Feed Plant

AEC will place its downtown St. Louis feed materials processing plant on standby beginning early in 1958, and consolidate its feed operations in the St. Louis area at the new plant at Weldon Spring, Mo., 33 mi W of St. Louis. Both plants are owned by AEC and operated by Mallinckrodt Chemical Works. A third plant, completed in 1953 and operated by National Lead Co. at Fernald, Ohio, will continue in production. Weldon Spring started operation in May this year. These feed materials plants process concentrated uranium ore to make pure natural uranium metal for the Hanford and Savannah River plutonium reactors, and UF_4 ("green salt") for the gaseous diffusion enrichment plants.

RCA Making Reactor Simulators

RCA is entering the reactor field as a maker of electronic simulators. RCA Service Co. has received a one-year AEC contract to build a reactor simulator under the Army Package Power Reactor program at Ft. Belvoir, where the machine will be used to train Defense personnel in reactor operation. E. L. Morrison Jr., formerly of Oak Ridge and Westinghouse's Atomic Power division, is project manager. RCA says it is seriously considering marketing reactor simulators commercially. The firm also recently obtained one of two contracts (along with Allis-Chalmers) to build the C-Stellarator at Princeton for the controlled fusion effort.

Radiation Effects Center Set Up by Air Force

An information center to gather and disseminate data on effects of radiation on systems and materials has been established at Battelle Memorial Institute by the Air Force. Intended primarily to aid the Aircraft Nuclear Propulsion program, the Radiation Effects Information Center's services will also be available to AEC, Army, Navy, and other government agencies. A 20-man team of experts headed by Battelle's C. B. Voldrich has been assigned to the Center.

AEC Dedicates Headquarters Building

President Eisenhower, AEC chairman Lewis L. Strauss and Congressional Joint Committee chairman Rep. Carl T. Durham laid the cornerstone of AEC's new $13.3-million headquarters building at Germantown, Md., 27 mi NW of downtown Washington. (NU, July '57, 23; *see also* p. 2). The cornerstone-laying was symbolic as the building is now finished; the staff will move in during the first half of January. The building is located at the junction of Routes 118 (Md.) and U. S. 240.

Dewey, General Mills Form Departments

Two more firms last month joined the roll of companies with nuclear departments. G. C. Dewey & Co., New York systems engineering and operations research firm, has opened a Department of Nuclear Science and Engineering to provide industry with analytical data in the reactor field, including calculations of neutron cross sections, radiation dosage and shielding, and reactor constants, flux, control, and critical mass. The firm has had several years' experience in the nuclear weapons program. . . . General Mills, well known in the nuclear industry for its master-slave manipulators, formed a Nuclear Equipment Department to make remote-control handling equipment and eventually other nuclear products.

AEC Opens Tokyo Office

W. H. Pennington will be AEC's atomic attaché in Tokyo, assisted by P. A. Roessler, AEC announced last month. A veteran of 14 years in nuclear energy work, Pennington, a chemist, will aid the rapidly-expanding Japanese nuclear development program, and will be available to other Far Eastern countries for consultation in the nuclear field.

Cambridge Accelerator Started

A two-handled spade was wielded by the presidents of Harvard and MIT to break ground for their joint project, the $6.5-million, 6-Bev Cambridge Electron Accelerator (NU, May '56, 24). It is to start operating in January 1960.

Lawrence Honored by AEC

The $50,000 Enrico Fermi Award for 1957 has been awarded by AEC to Ernest O. Lawrence, of the Univ. of California and director of its Radiation Laboratory. Lawrence helped open the way for modern development of nuclear physics with his invention, back in 1929, of the cyclotron. He is the second Fermi Award winner, the first last year having gone to John von Neumann.

AEC Signs for $10-Million U-Mill

Homestake Mining Co. has signed a $10-million contract with AEC for construction of a U-ore processing mill near Grants, N. M. Utah Construction Co. of San Francisco, now building a similar mill near Riverton, Wyo., for Lucky Mc Uranium Corp. (NU, Dec. '57, R3), will build the 1,500-ton/day capacity mill for Homestake-Sapin Partners.

Oklahoma State Starts Its Reactor

A nuclear engineering symposium marked the dedication of Oklahoma State Univ.'s AGN-201 training reactor at Stillwater, Okla.

Fiat Seeking U. S. Financing

It is authoritatively reported in Rome that Fiat plans to market $5-million worth of company shares in the U. S. to help finance its nuclear activities. The firm is negotiating with a group of American banks that would sell the stock and act as guarantors. Further sales are contemplated if the initial one is successful.

AEC Financial Report Shows Greater Spending

AEC's financial report for fiscal 1957, second such report to be published (NU, Jan. '57, R3), showed that total cost of Commission operations had risen to $1.968-billion, from $1.607-billion in fiscal 1956.

The $2-billion spent in round figures for operations went for these items: production of fissionable materials, $788-million (up 8% over '56); uranium ore, $402-million (deliveries up 43%); weapons, $337-million (up 20%); reactor development, $276-million (up 62%, from $170-million in '56); chemistry, physics, and metallurgical research, $59-million (up 16%); cancer, medicine and biology research, $33-million (up 10%); and administrative, community and other expenses, $73-million (up 42%).

Investment in plant owned by AEC rose $0.2-billion, from $6.7- to $6.9-billion. Plant completed during the year included production reactor facilities at Hanford and Savannah River, $69-million; gaseous diffusion facilities at Oak Ridge, Paducah and Portsmouth, $47-million; feed materials facilities at Weldon Spring, Mo., and Fernald, $45-million; and "various" research facilities of $46-million and weapons facilities of $43-million. The additions were partially offset by plant retirements of $87-million, including $21-million for the *Seawolf* land-based prototype at West Milton, N. Y.

Experimental power reactor research and development totaled $56.7-million, up 24% over '56; construction costs were $33.3-million, compared to $9.4-million in '56.

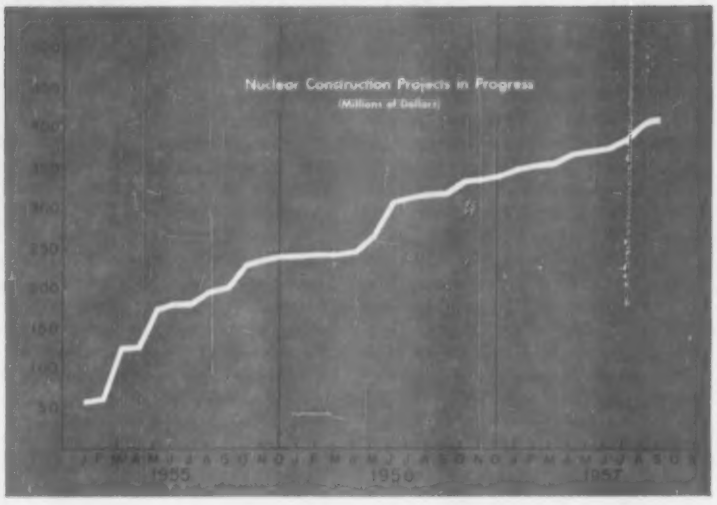
Aircraft reactor research and development costs were up 67%, from $52-million in '56 to $86.7-million; and construction costs went up 158%, from $2.4- to $6.2-million. The total cost of $92.9-million represents a 257% increase over the $25.9-million spent two years ago, fiscal '55.

Naval propulsion reactor research and development costs in '57 totaled $79.1-million, a 63% increase over '56. Construction costs at $26-million were up 165%.

The controlled fusion program cost $11.1-million in '57, up 57%; construction costs on Project Sherwood were $495-thousand, as compared with $141-thousand in '56.

Cost/lb of uranium concentrates delivered from U. S. sources dropped 10% (from $11.96 to $10.80/lb).

NUCLEONICS Statistics of the Month



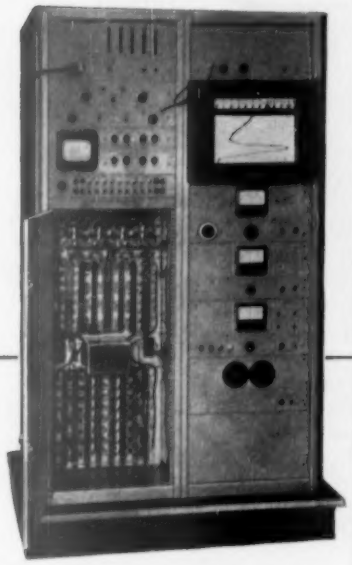
	Latest month	Month ago	Year ago
Nuclear Contracts*—			
Contracts awarded for federal projects ($10⁶)	9,583	10,549	8,842
Proposed construction, privately-owned ($10⁶)	0	0	500
Contracts awarded, private work ($10⁶)	9,500	1,750	6,400
Backlog of private projects ($10⁶)	226	226	167
Access Permits†—			
Access permits issued	25	18	42
Total access permit holders	1,381	1,372	1,075
Isotope Use†—			
Applications for isotope use	344	366	379
Cumulative total of isotope users	4,353	4,298	3,559
Oak Ridge National Laboratory shipments	1,201	1,106	1,216
Public and private export shipments	407	94	238
Employment†—			
AEC employment	6,852	6,853	6,601
Construction and design contractors' employment	11,581	12,357	14,810
Total operating contractors' employment	99,150	99,477	91,897
Production workers	50,577	50,619	49,702
Research and development employees	41,822	42,296	36,123
Miscellaneous workers	6,751	6,562	6,072

	Operating	Building	Contracted
U. S.-built Reactors‡—			
Power, domestic	1	5	8
Power, for export	0	1	5
Power experiments	5	4	3
Military and naval	5	20	10
Research and test, domestic	35	21	23
Research and test, for export	6	14	4
Foreign-built Reactors‡—			
Power, domestic	4	24	12
Power, for export	0	0	2
Research and test, domestic	24	17	6
Research and test, for export	2	10	1

* From *Construction Daily*, a McGraw-Hill periodical † From AEC ‡ NUCLEONICS figures

NUCLEAR DATA PROCESSING — FROM DETECTOR TO READOUT

Multi-Channel Analyzers

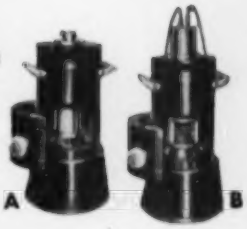


Radiation, no matter what its source, can best be analyzed by RCL's 256-channel analyzer. Speed, precision, reliability are but a few of the qualities that make RCL the leader in the spectrometry field. Also available, 1024-channel Neutron Time of Flight Analyzer. See the new RCL catalogue for more information.

Fast-Slow Neutron Survey Meter

Highest Gamma discrimination, transistorized, lighter weight, printed circuitry, log scale, and reduced power requirements are some of the features that make the RCL survey meter the most advanced meter available. See RCL's new catalogue for all of your instrument requirements.

Scintillation Detectors

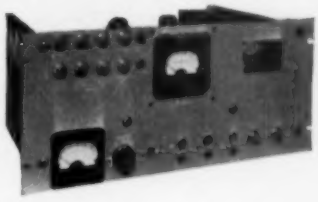


New scintillation detectors are now available from RCL. Illustrated above is the flexibility when used with RCL's new dual purpose shield. (a) with a well type crystal for test tube samples and (b) with a solid crystal for flat samples. Alpha, beta and gamma crystals available, as well as other shields and collimators. See RCL's new catalogue.

And now—RCLiac 32 and RCLiac 128 Amazing new scaler-analyzers

Combination scaler-analyzers handle every operation from gross counting through scintillation-spectroscopy and pulse height analysis. Special feature: cathode ray tube displays numbers or curve. Six transistorized, plug-in modules include display, power supply, memory, amplifier, control and arithmetic circuits.

Reactor Controls

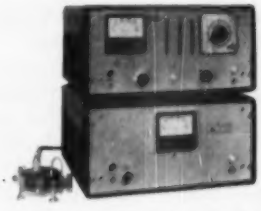


The Composite Safety Amplifier illustrated above is only one of the many reactor control instruments manufactured by RCL. The high quality and reliability that is mandatory with these items is guaranteed by RCL.

Sensing Elements

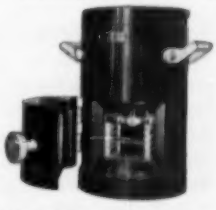
The Mica Window Counter above is representative of the most complete line of sensing elements available in the nuclear field today. Scintillation detectors, GM counters, fast and slow neutron detectors, ion chambers, and flow counters are outlined in RCL's new 132-page catalogue, available immediately. Use the coupon below to send for yours today.

Scalers and Proportional Counting Systems



The system shown above is a complete gross counting system for alpha, beta and gamma radiations. It consists of a decade scaler, a high voltage supply with pulse amplifier, and a preflush flow detector. Operates with GM, scintillation and proportional detectors.

Counting Room Accessories



This Vertical Iron Shield is representative of RCL's complete line of counting room accessories. Other shields, safety devices, sample handling items and accessories demonstrate RCL's complete line of nuclear equipment and instruments. Send for RCL's new catalogue for a complete list of products.

Send for RCL's NEW 132-page catalogue TODAY.

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Please rush RCL's new 132-page catalogue to me immediately.

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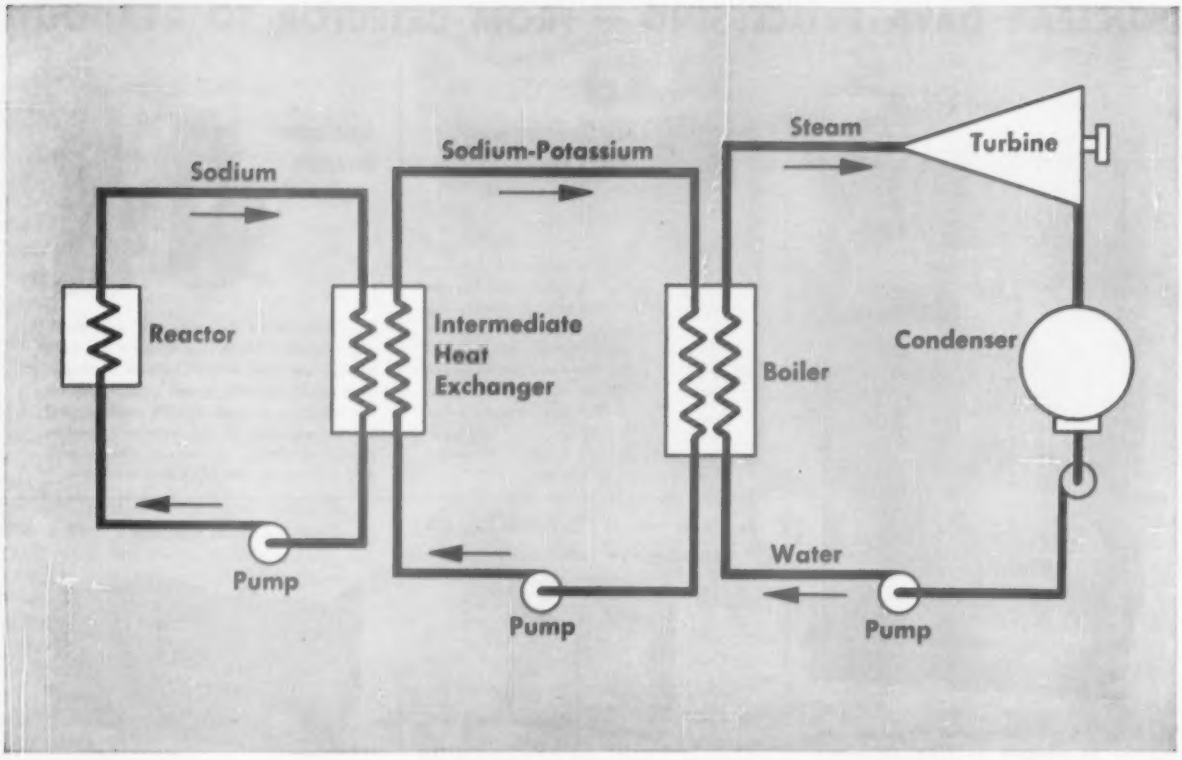


What makes LIQUID SODIUM a first-rate Reactor Coolant?

At least 6 nuclear power plants now completed, planned, or under construction will use sodium metal in liquid form to transfer heat from reactor to steam generation circuit. Why have the designers of these systems selected sodium for the job?

Because sodium is a liquid from 207° to 1621° F., a sodium-cooled system can operate at high temperatures without expensive pressurization. In this way it can yield steam cycle efficiencies equivalent to those of conventional power plants.

Liquid sodium can be pumped and circulated as easily as water. Further, because liquid sodium has good electrical conductivity, it can be circulated with an electromagnetic pump containing no moving parts. With the low melting point of 207° F., it won't solidify in the heat exchanger system if the reactor is slowed down.

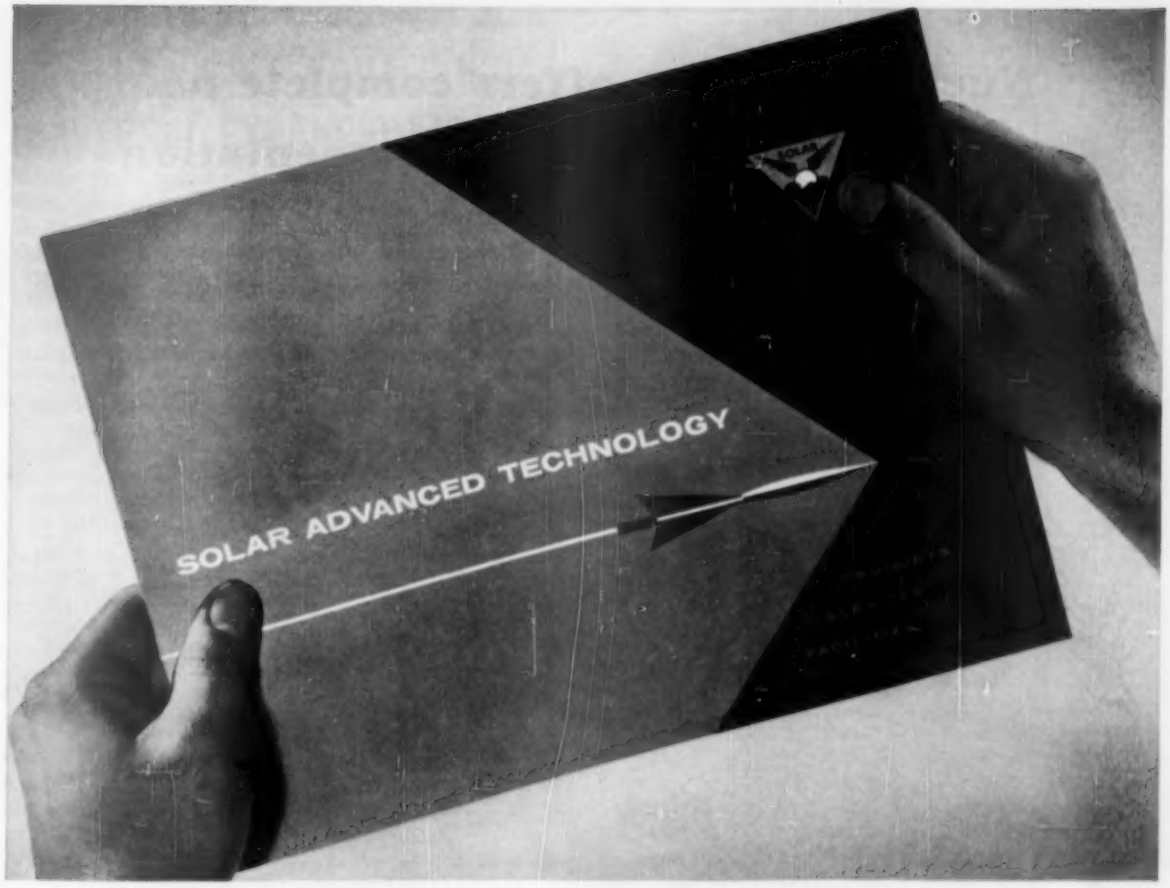
Sodium has high heat conductivity, as well as good radiation stability and high temperature stability. In the absence of oxygen, water, and other impurities, it can be handled without corrosion in iron and steel equipment.

U.S.I. supplies more sodium to outside industry than any other producer of the metal. For complete information on sodium and its handling, send for the free 40-page booklet "HANDLING METALLIC SODIUM ON A PLANT SCALE".

SOME REACTORS WHICH WILL USE SODIUM:

(1) **California**—sodium-graphite reactor, 7,500 kw
(2) **Nebraska**—sodium-graphite reactor, 75,000 kw
(3) **Idaho**—fast breeder reactor, 15,000 kw
(4) **Michigan**—fast breeder reactor, 100,000 kw
(5) **Alaska (Anchorage)**—sodium-deuterium reactor, 10,000 kw
(6) **Seawolf**—sodium-beryllium reactor
(7) **New York**—sodium-beryllium reactor

U.S.I. INDUSTRIAL CHEMICALS CO.
Division of National Distillers and Chemical Corporation
99 Park Avenue, New York 16, N. Y.



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NUCLEAR AND MISSILE DEVELOPMENT at Solar is centered in a team of qualified scientists devoted entirely to nuclear and guided missile technology. This experienced group represents only one example of Solar's advanced planning for today—and tomorrow.

For three decades the company has been an acknowledged leader in introducing new products and new production techniques. Products range from powerful gas turbines to industrial expansion joints, from missile components to thrust-producing afterburners, from stainless steel aircraft nacelles to "hot parts" for advanced jet engines.

A new 16-page brochure giving full details about Solar's products, services and facilities is available upon request. Send for it today and learn how Solar can put this experience to work for you. Write to Dept. D-204, Solar Aircraft Company, San Diego 12, California.

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Nuclear-Chicago offers complete new nucleonic laboratory instrumentation

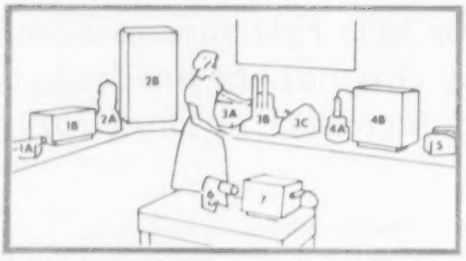
More than two years' research went into the 33 *all new* instruments that have been added to the Nuclear-Chicago line now offered to the small or large nucleonic research laboratory. Among these high-quality advance-design instruments are many completely automatic counting systems.

Whether you are just starting a modest radioisotope laboratory or expanding present facilities, call on Nuclear-Chicago. For complete details on the new Nuclear-Chicago line, write for our new Catalog "Q".

1A Model DS5-3P Scintillation Detector with exposed 2" x 2" sodium iodide crystal and **1B** Model 132 Analyzer Computer for precision gamma-ray measurements using pulse-height discrimination techniques.

2A Model DS5-5 Scintillation Well Counter with exclusive "scaler-spectrometer" circuit and **2B** Model 1820 Recording Spectrometer for automatic quantitative energy separation of gamma-ray spectra.

3A Model 183B Count-O-Matic Binary Scaler, **3B** Model C110B Automatic Sample Changer with Model D47 Gas Flow Counter, and **3C** Model C111B Printing Timer for completely automatic changing, counting, and recording of as many as 35 soft beta emitting radioactive samples.



4A Model 3054 Manual Sample Changer with Model DS5 Scintillation Detector which features interchangeable alpha, beta, or gamma sensitive crystals connected to Nuclear-Chicago's finest scaler, the **4B** Model 192A Ultrascaler. Model 192A features decade scale of 10,000, one millivolt sensitivity, and precision automatic circuitry.

5 Model 2612P Portable Survey Meter contains a 1.4 mg/cm² thin window G-M tube for surveying for alpha, beta, or gamma contamination up to 20 milliroentgens per hour.

6 Model 2586 "Cutie Pie" features interchangeable ionization chambers for measuring beta, gamma, or x-radiation up to 250 roentgens per hour.

7 Model 1620A-S Analytical Count Rate Meter offers a wide choice of full scale ranges, four time constants, wide range high voltage supply. It is shown with a Model D34 thin window G-M tube and P11 probe for continuous monitoring or analytical radioactivity determinations.

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ZALLEA SELF EQUALIZING UNIVERSAL EXPANSION JOINTS

Zallea Self Equalizing Universal Expansion Joints absorb any movement in any direction . . . axially, laterally, angularly, or any combination of these . . . making them ideal for protecting process vessels and rotating equipment from thermal expansion of connected piping.

When there is a leg or 90° offset to a long run of pipe, one Zallea Universal Expansion Joint will frequently take care of the entire system.

Under conditions of low load requirement for anchors or equipment . . . or when compressive loading in pipe walls must be kept to a minimum . . . Zallea Universal Expansion Joints are the most economical answer to the problem. They absorb the internal pressure thrust. The piping system need only carry the relatively small deflection loads.

Many types of Zallea Tied Expansion Joints are designed especially to absorb lateral deflection or rotation when there is pressure in the line. Universal, Modified Universal, Hinged, Gimbal, Pressure Balanced and Universal Pressure Balanced Expansion Joints give you a choice of the type best suited to your particular application.

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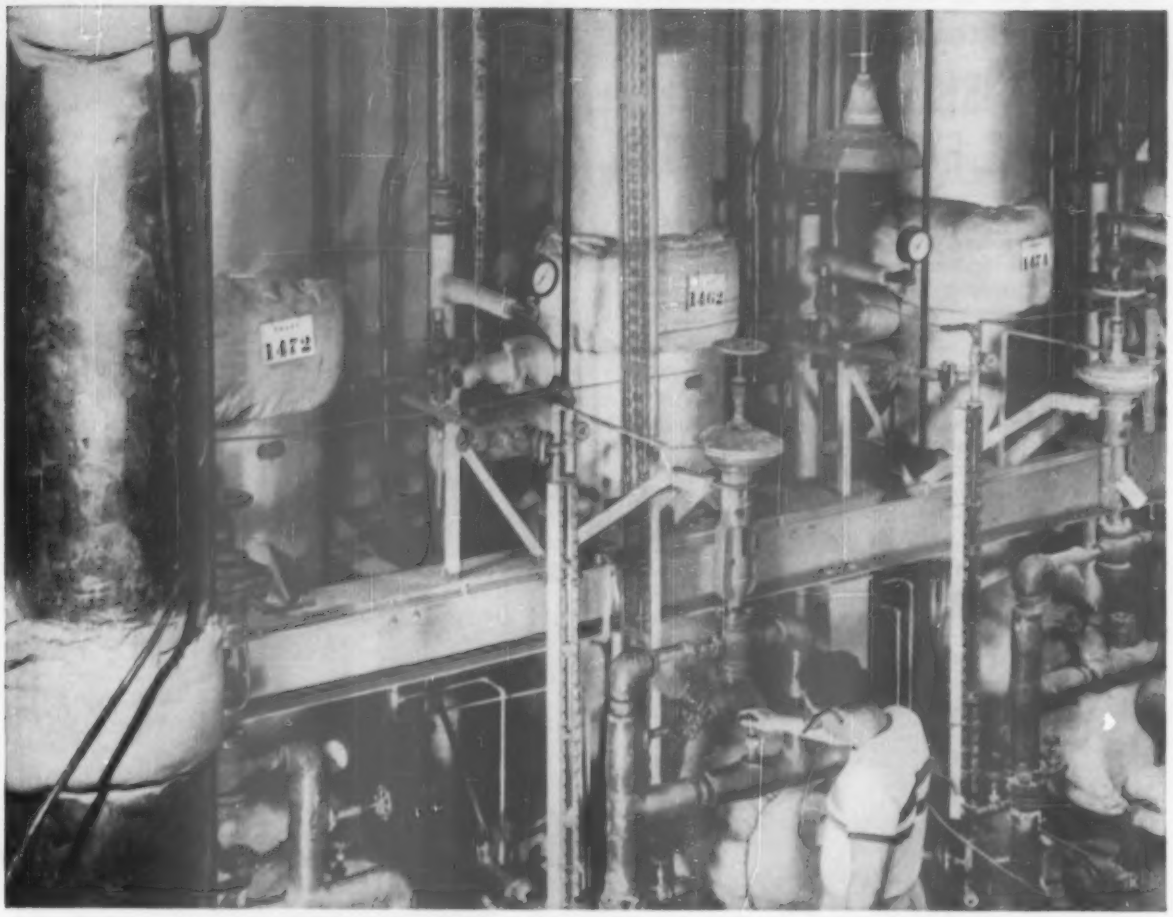
World's largest manufacturer of expansion joints

24" dia. Zallea Universal Self Equalizing Expansion Joints in service lines of the engine test facility of a West Coast aircraft manufacturer.



Monel nickel-copper alloy withstands corrosive fluoride attack in these series-connected towers for separating B-10 and B-11 isotopes found in natural boron.

Hooker "divorces" boron isotopes in unusual 350-foot, 6-tower, Monel fractionator

...gets 3 pounds per day of precious neutron-absorbing boron-10

The towers shown above are part of a six-tower fractionating system that may be the world's most effective. It is located in the AEC's Boron-10 plant, designed and constructed by Singmaster & Breyer and operated by Hooker Electrochemical Co.

In 350 feet of total height (they're connected in series) they enrich a dimethyl ether-boron trifluoride complex containing 18.8% B-10 to one containing 92% of this isotope.

Strangely, the separation of the B-10 and B-11 constituents does not depend on volatility. Instead it depends on a difference in the rates of association and dissociation of the DME-BF_3 complex. The lighter B-10 compound comes off in the liquid bottoms and the heavier B-11 compound in the vapor tops.

Corrosion potential is high

With fluorides at 105°C., and with the inevitable trace moisture present in a system of this type, a high corrosion potential is established.

For this reason, from the point where boron trifluoride enters the system via a Monel nickel-copper alloy reducing valve, most equipment is made of Monel* nickel-copper alloy ... tanks, columns, re-boilers, condensers, pumps, valves, piping.

Columns are filled with protruded Monel alloy packing

Monel alloy is highly resistant to the fluoride corrosion that is so often a problem in preparing isotopes and nuclear fuels. What's more, it has the high strength and fabricating properties needed for pressure vessels and allied equipment.

For technical assistance with your own corrosion problems, call on Inco's Development and Research Division.

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INCO NICKEL ALLOYS

NUCLEONICS DECEMBER, 1957

Jerome D. Luntz, *Editor*

Fusion: International Perspective

AT THIS WRITING, the American public is in a state of intellectual and emotional shock over the Russian advances in satellites and missiles. The President of the United States has felt impelled to go before the public to allay its fears about where we stand vis-a-vis the U. S. S. R. Assurances have been given by various officials that we're not really behind in science and technology and that being "first" is not what is significant.

Whether or not these last two assurances are valid, the most important effect of *sputnik* and its companions has been to make apparent the need for a soul-searching analysis of this nation's role in science and technology. Perhaps more significant is the fact that there has come to light, really for the first time, the need to do this from the perspective of the international implications of decisions on domestic activities. We cannot develop domestic policies without understanding their potential repercussions abroad.

At this very moment, the U. S. nuclear power program is undergoing a thorough and critical analysis of whether it is designed to meet whatever foreign policy requirements this nation has.

Unfortunately, the pressure that is mounting for examining both our power reactor effort and our broad program in science and technology is primarily from the short-term point of view, the desire to out-*sputnik* the Soviets and to do it quickly. What we need instead is to immerse ourselves in the lessons of the day and to come out with a sophisticated plan for a look-ahead at the next decade, not just 1958 or 1959.

As an example, an ideal example perhaps, there is the field of controlled thermonuclear power. This is a field that is in its infancy, that is being pioneered right now. But although fusion efforts are in not much more than the test tube stage, there is little question that the technical feasibility of a fusion reactor will be proved and that at some later date an economic system will be developed. These achievements will bring with them an important new source of energy for the world.

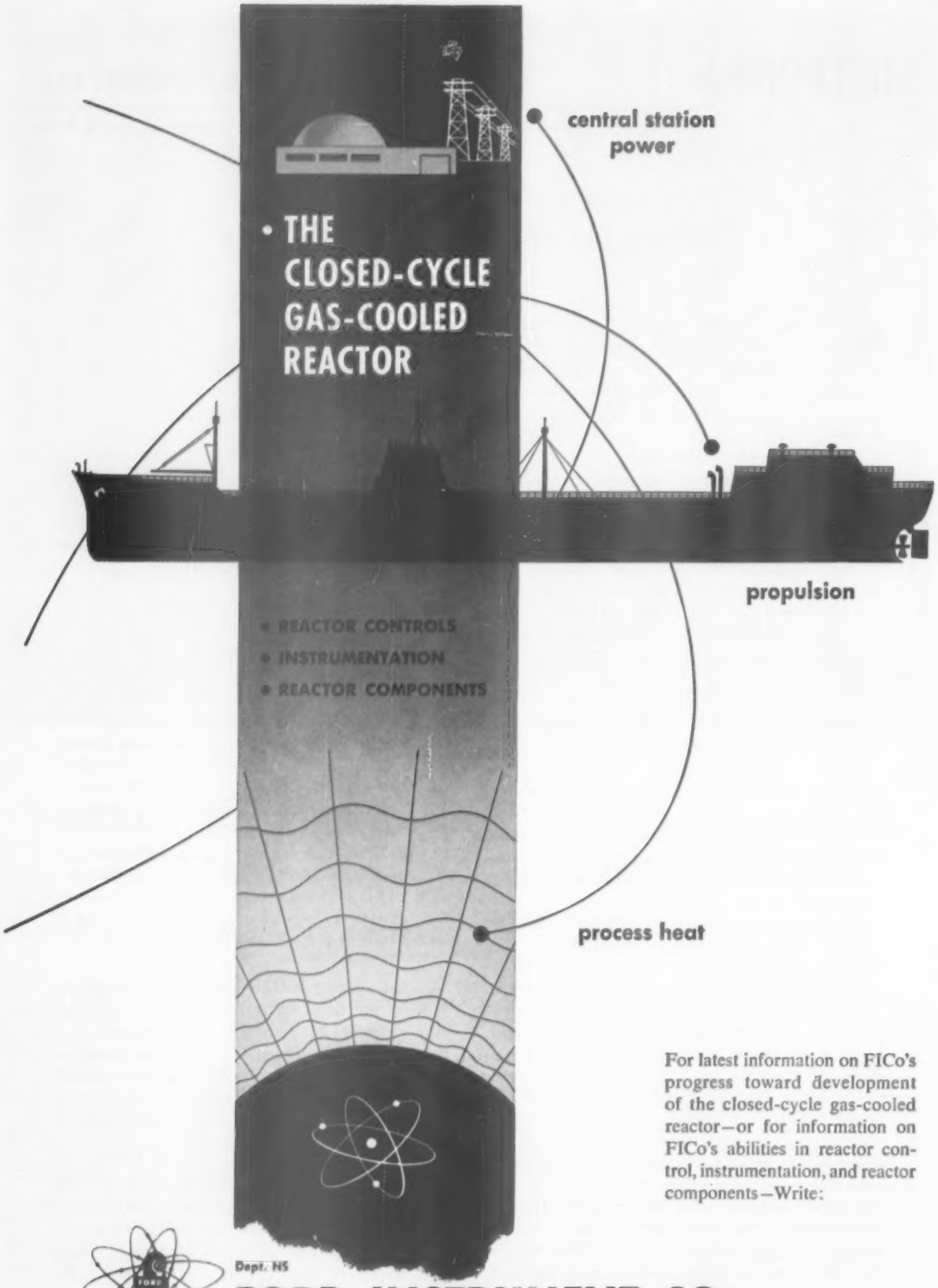
In light of this, the question we would like to pose is whether the U. S. is doing everything it should be doing and can be doing in pressing the fusion program. This is another area in which we're pitted against the Russians and also the British. And some responsible scientists in the U. S. program feel we're behind the U. K. and that we may even be behind the U. S. S. R.

Here is a field where success lies in the future. There is ample time, if a decision is made soon enough, for the U. S. to mount an aggressive attack on the major technological problems and, with good hope for success, to demonstrate to the world by action and accomplishment that the U. S. is, in fact, the world leader in developing significant peaceful uses of science and technology.

Instead, we appear to be moving along at a very leisurely pace in a rather modest program. We are doing this behind closed doors in a way that is stifling broad interest in the work among many who might make useful technical contributions. The secrecy is also hurting those actually involved in the project, not only because of the welter of problems that arise under a system of classification, but also because of the absence of constructive scientific criticism from the outside. It is perhaps a truism to say that science flourishes in a free and open atmosphere, but it seems to bear restatement time and again.

As the country in the world that has the greatest potential for technological competence, it is criminal for us not to look beyond our own borders at the needs of the rest of the world, a world which, if only for selfish reasons we should want to help. Because we have no urgent need for a new source of energy is no reason for us to ignore the wants of other nations.

Thus, in the case of controlled fusion, we should do everything possible NOW to see that the U. S. is the world leader five, ten, fifteen years from now, or whenever success is achieved. Certainly stepping up our effort and shedding the wraps of secrecy are important to accomplishing this.



For latest information on FICo's progress toward development of the closed-cycle gas-cooled reactor—or for information on FICo's abilities in reactor control, instrumentation, and reactor components—Write:

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FORD INSTRUMENT CO.

DIVISION OF SPERRY RAND CORPORATION • 31-10 THOMSON AVENUE, LONG ISLAND CITY 1, N. Y.

Characteristics of SRE

Location:	Santa Susana, California
Owner:	Atomic Energy Commission
Prime contractor:	Atomics International (AI)
On line:	July 25, 1957
Output—heat:	20,000 kw
—electrical:	6,000 kw

Component Design Data

Core tank

Height:	19 ft
Inside diameter:	11 ft
Wall thickness:	1½ in.
Composition:	stainless steel (304)

Core

Configuration:	cylinder
Size:	6 ft dia × 6 ft
Fuel load—U^{235}:	62.5 kg
—U^{238}:	2,190 kg
Composition	
—graphite:	86.5 vol%
—sodium:	6.6 vol%
—uranium:	3.0 vol%
—zirconium:	1.7 vol%
—stainless steel:	0.32 vol%
—NaK:	0.08 vol%
—voids:	1.8 vol%

Control and safety rods

Configuration:	cylinder
Number	
—shim:	2
—regulating:	2
—safety:	4
Poison:	B-Ni
Total rod worth:	~9.6% $\Delta k/k$
Weight	
—shim:	56 lb
—regulating:	56 lb
—safety:	22 lb
Withdrawal rate	
—shim:	3.6 in./min
—regulating:	36 in./min
—safety:	36 in./min
Scram time:	0.6 sec

Rod drives

Type:	lead screw and nut
Number:	8
Power requirement	
—shim:	220 v, 3 ϕ
—regulating:	120 v, 2 ϕ
—safety:	220 v, 3 ϕ
Motor rating	
—shim:	1,140 rpm, ¼ hp
—regulating:	3,600 rpm, ⅛ hp
—safety:	1,425 rpm, ½ hp
Position indicator accuracy	
—mechanical:	±1/16 in.
—selsyn:	±½ in.

Fuel elements

Type:	slug
Number:	43
Rods per element:	7
Slugs per rod:	12
Slug dimensions:	¾ in. dia × 6 in.
Slug temp.—avg:	1,000° F
—max:	1,200° F
Burnup:	2,500 Mwd/ton

Nuclear Design Data

Moderator

Type:	graphite
Average neutron energy:	0.06 ev
Thermal-neutron flux—avg:	3.5×10^{13} n/cm²/sec
—max:	5×10^{13} n/cm²/sec
Prompt-neutron lifetime:	0.5×10^{-3} sec
Eff. delayed-neutron fraction:	0.0075
Prompt temp. coefficient:	-0.78×10^{-5}/° F*
Endurance:	9,600 FPH

Control requirement

Burnup compensation:	1.4% $\Delta k/k$
Peak xenon override:	2.4% $\Delta k/k$
Equilibrium samarium:	0.6% $\Delta k/k$
Total control available:	4.4% $\Delta k/k$

Heat Transfer Data

Coolant

Type:	sodium
Volume in primary system:	26,000 liters
Volume in secondary system:	8,300 liters
Flow:	1,180 gpm
Temperature	
—inlet:	500° F
—outlet:	960° F
Pressure:	3 psig

Heat flux

Max/avg ratio	
—axial:	1.35
—radial:	1.63
Core—average:	209,000 Btu/ft²/hr
—maximum:	340,000 Btu/ft²/hr
—burnout:	900,000 Btu/ft²/hr

* At 400° F. Steady-state coefficient is positive ($\sim 3 \times 10^{-5}$), becomes negative at ~700° F (average reactor temperature at full power is 730° F). No control is required to compensate for temperature coefficient because it is positive.

NUCLEONICS REACTOR FILE... NO. 3

Reactors on the Line

SRE

Sodium Reactor Experiment

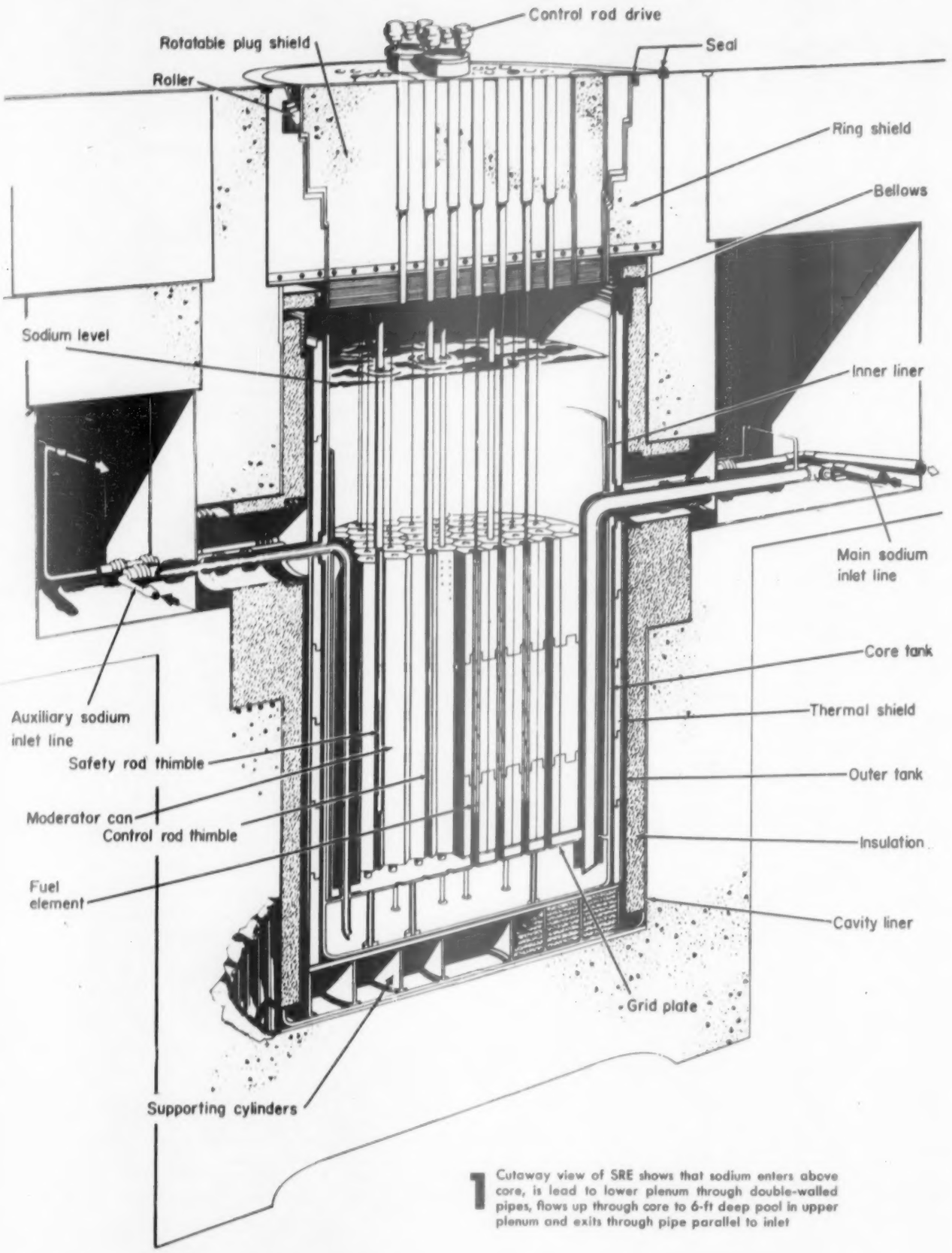
tear along perforation

This foldout is removable

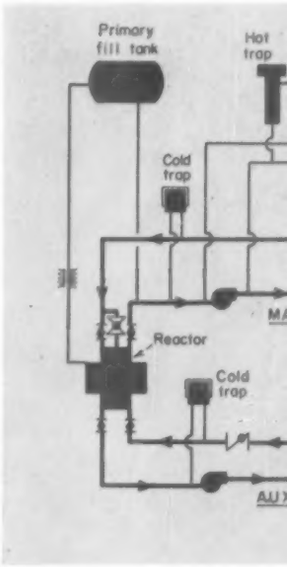
SRE on

SRE went critical on April
These were low-power runs.
The reactor is not scheduled
rather slow approach to full p
mental, high-temperature rea
(460° F at full power). As
need to be performed at low-p

Among these tests are meas
and transient conditions to e
of the temperature of the sodi
sizes of orifice plates to be
effectiveness of the hot and col



1 Cutaway view of SRE shows that sodium enters above core, is lead to lower plenum through double-walled pipes, flows up through core to 6-ft deep pool in upper plenum and exits through pipe parallel to inlet



2 Main and auxiliary sodium circuits include radioactive primary and nonradioactive secondary loops. Auxiliary circuit provides circulation to remove residual heat in event of scram caused by failure of main circuit

3 Core structure consists of 119 moderator and reflector cans. Moderator cans have center channels for fuel elements. Long tubes projecting upward are vents for off-gas from graphite in cans

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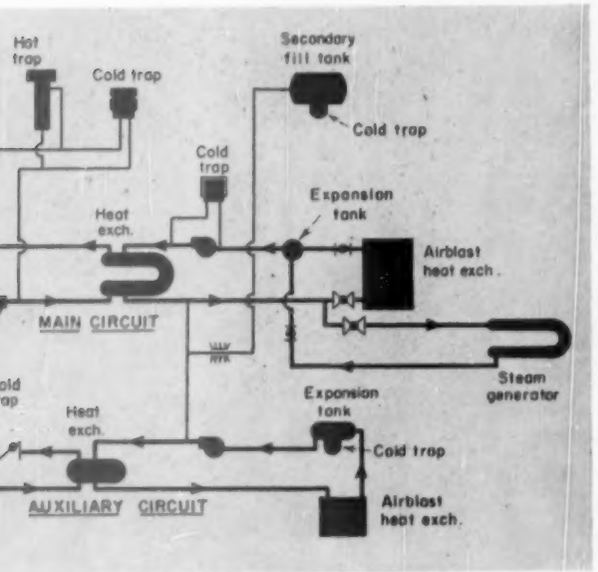
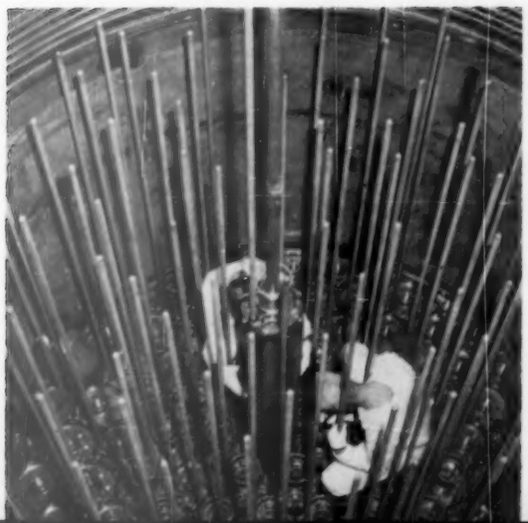
n the Line

April 25, 1957, and generated electric power on July 25 and 26.
runs. There were further low-power runs during November.
eduled to operate at full power until early next year. This
full power is attributed to the fact that the SRE is an experi-
e reactor with a large temperature difference across the core
As a result, a program of tests and analyses of performance
low-power levels.
measurements of a variety of temperatures under steady-state
s to enable the calculation of thermal stresses, measurements
e sodium as it leaves the fuel channels to determine the proper
be installed on the fuel elements, and measurements of the
nd cold traps in reducing the oxygen content of the sodium.





Characteristics of SRE

Location: Santa Susana, California
Owner: Atomic Energy Commission
Prime contractor: Atomics International (AI)
On line: July 25, 1957
Output—heat: 20,000 kw
—electrical: 6,000 kw

Component Design Data

Core tank

Height:	19 ft
Inside diameter:	11 ft
Wall thickness:	1½ in.
Composition:	stainless steel (304)

Core

Configuration:	cylinder
Size:	6 ft dia × 6 ft
Fuel load—U^{235}:	62.5 kg
—U^{238}:	2,190 kg
Composition	
—graphite:	86.5 vol%
—sodium:	6.6 vol%
—uranium:	3.0 vol%
—zirconium:	1.7 vol%
—stainless steel:	0.32 vol%
—NaK:	0.08 vol%
—voids:	1.8 vol%

Control and safety rods

Configuration:	cylinder
Number	
—shim:	2
—regulating:	2
—safety:	4
Poison:	B-Ni
Total rod worth:	~9.6% $\Delta k/k$
Weight	
—shim:	56 lb
—regulating:	56 lb
—safety:	22 lb
Withdrawal rate	
—shim:	3.6 in./min
—regulating:	36 in./min
—safety:	36 in./min
Scram time:	0.6 sec

Rod drives

Type:	lead screw and nut
Number:	8
Power requirement	
—shim:	220 v, 3 ϕ
—regulating:	120 v, 2 ϕ
—safety:	220 v, 3 ϕ
Motor rating	
—shim:	1,140 rpm, ¼ hp
—regulating:	3,600 rpm, ⅛ hp
—safety:	1,425 rpm, ½ hp
Position indicator accuracy	
—mechanical:	±1/16 in.
—selsyn:	±½ in.

Fuel elements

Type:	slug
Number:	43
Rods per element:	7
Slugs per rod:	12
Slug dimensions:	¾ in. dia × 6 in.
Slug temp.—avg:	1,000° F
—max:	1,200° F
Burnup:	2,500 Mwd/ton

Nuclear Design Data

Moderator

Type:	graphite
Average neutron energy:	0.06 ev
Thermal-neutron flux—avg:	3.5×10^{13} n/cm²/sec
—max:	5×10^{13} n/cm²/sec
Prompt-neutron lifetime:	0.5×10^{-4} sec
Eff. delayed-neutron fraction:	0.0075
Prompt temp. coefficient:	-0.78×10^{-5}/° F*
Endurance:	9,600 FPH

Control requirement

Burnup compensation:	1.4% $\Delta k/k$
Peak xenon override:	2.4% $\Delta k/k$
Equilibrium samarium:	0.6% $\Delta k/k$
Total control available:	4.4% $\Delta k/k$

Heat Transfer Data

Coolant

Type:	sodium
Volume in primary system:	26,000 liters
Volume in secondary system:	8,300 liters
Flow:	1,180 gpm
Temperature	
—inlet:	500° F
—outlet:	960° F
Pressure:	3 psig

Heat flux

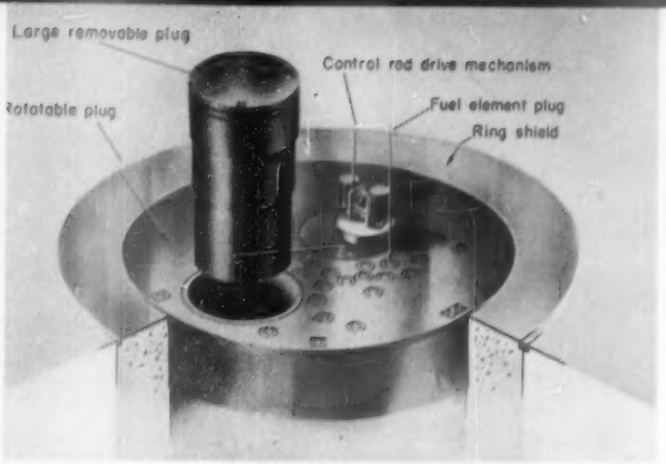
Max/avg ratio	
—axial:	1.35
—radial:	1.63
Core—average:	209,000 Btu/ft²/hr
—maximum:	340,000 Btu/ft²/hr
—burnout:	900,000 Btu/ft²/hr

* At 400° F. Steady-state coefficient is positive ($\sim 3 \times 10^{-5}$), becomes negative at ~700° F (average reactor temperature at full power is 730° F). No control is required to compensate for temperature coefficient because it is positive.

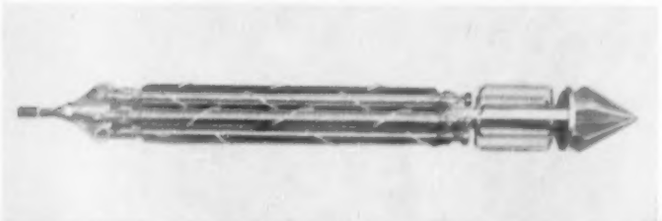
Reactors on the Line

SRE

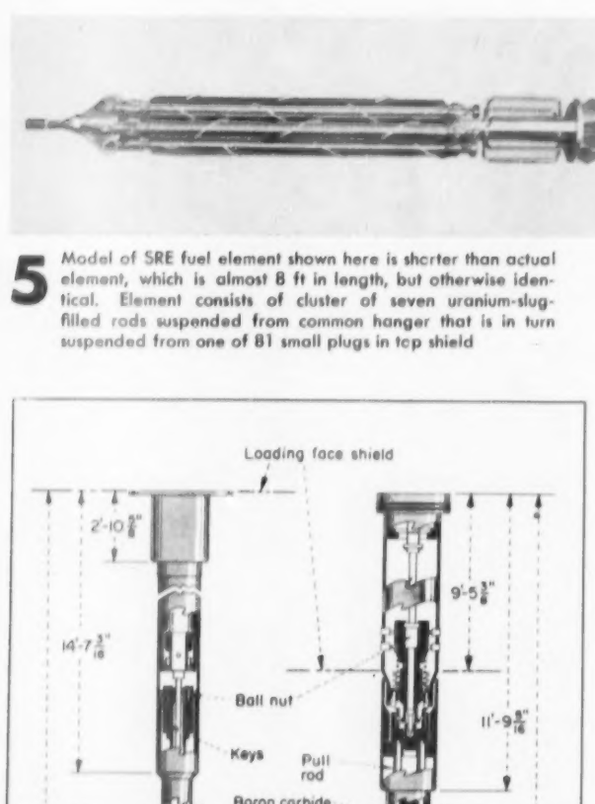
Sodium Reactor Experiment



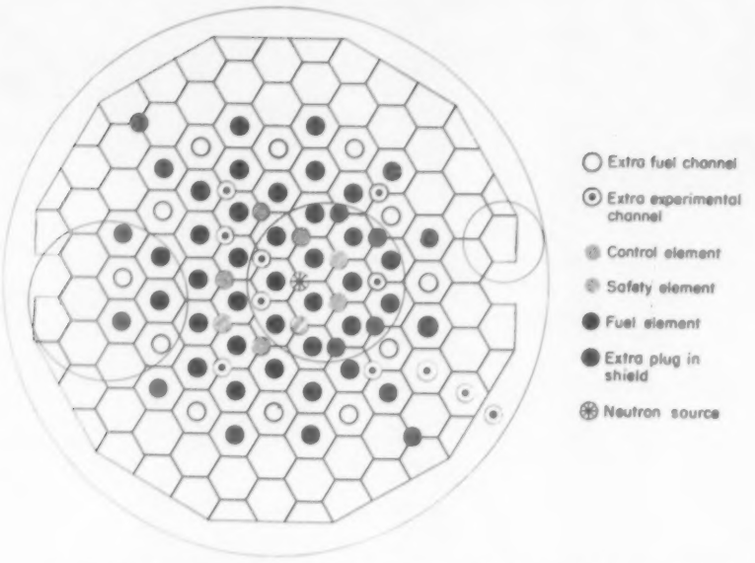
4 Drawing of top shield shows one of large removable plugs provided so as to be able to remove moderator and reflector cans from core. Stationary ring shield, rotatable plug shield, and large removable plugs are made of stainless steel encased concrete



5 Model of SRE fuel element shown here is shorter than actual element, which is almost 8 ft in length, but otherwise identical. Element consists of cluster of seven uranium-slug-filled rods suspended from common hanger that is in turn suspended from one of 81 small plugs in top shield

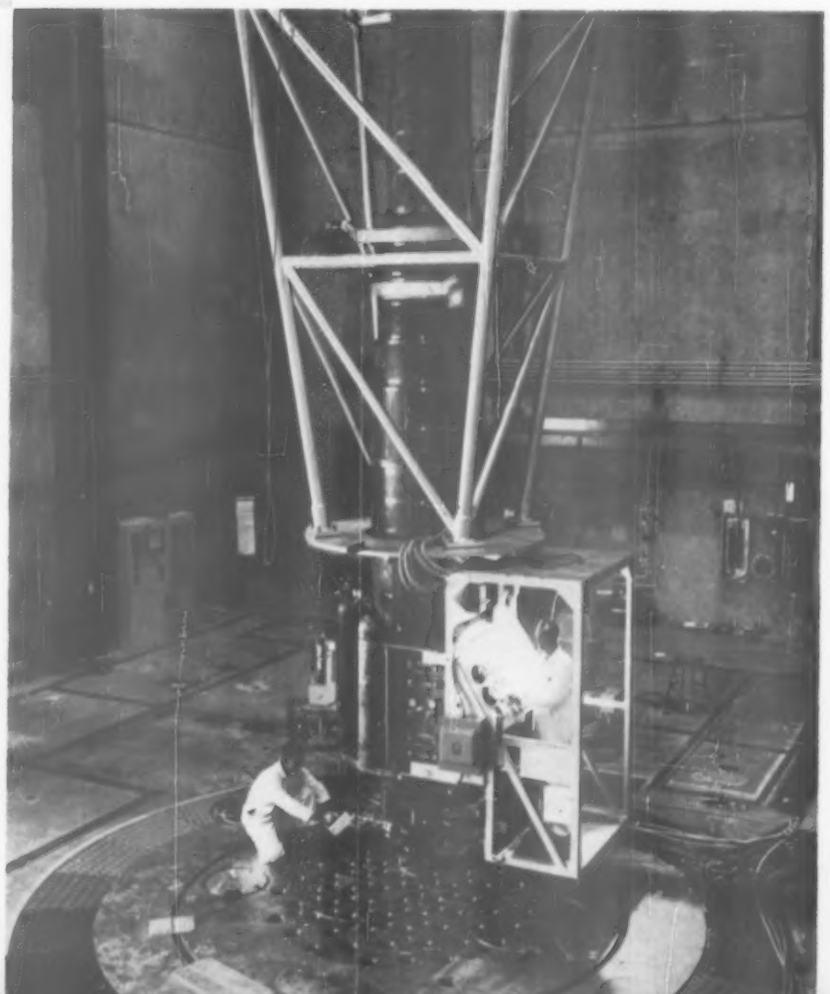


6 Cutaway view of shim rod (left) and regulating rod (right) shows that column of B-Ni poison rings assembled on pull tube is moved by lead screw and nut arrangement inside thimble



7 Loading pattern of SRE core shows position of fuel elements in center channels of hexagonal moderator cans, control and safety rods in corner channels, etc. Top shield contains small plugs that correspond to each position and some extras

8 Fuel handling cask, in position over reactor top shield, is suspended from bridge crane. Cask is 34 ft high, weighs 65 tons, including 35 tons of lead shielding, and has room for two fuel elements



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escription of SRE

lium Reactor Experiment (SRE) is a thermal, ooled reactor using graphite as the moderator and nriched uranium as the fuel.

Liner, Core Tank, and Top Shield

ntire reactor is located below grade, with the upper f the top shield at floor level in the reactor build- e cavity liner sits on a 4-ft thick reinforced con- l at the bottom of the excavation. It is fabricated -carbon steel 1 in. thick at the bottom and ¼ in. the sides. It served as the form about which the k reinforced concrete foundation was poured onstruction; during operation it provides means of g heat from this region. The maximum estimated duction in the concrete is 50 Btu/ft²/hr. This is moved by toluene circulated through 28 standard l pipes tack-welded to the outside of the liner.

rt for the core tank is provided by four concentric 21 in. high in the bottom of the cavity liner. tank is 19 ft high and 11 ft in diameter. It is d of 1½-in. stainless-steel plate. Between the k and the cavity liner is a steel thermal shield hick, an outer tank 19 ft high and 12½ ft in diam- 12 in. of insulation (calcined diatomacious silica stos coated with sodium silicate). The thermal fabricated from low-carbon steel in the form of a seven rings with interlocking joints. The rings welded to each other, and the bottom ring simply the bottom of the outer tank.

uter tank is provided to serve as an emergency containing the sodium in the event a leak should in the core tank. Its bottom is 2½ in. thick and are ¼ in. thick. It is fabricated of low-alloy steel. p of both the core and outer tanks are bellows to hermal expansion and to provide a gas seal for the tmosphere that exists above the level of the sodium re tank and in the annulus between the core tank uter tank.

p shield consists of two components—a stationary d supported on a ledge of the cavity liner and a plug shield supported by steps on the inside of hield. The ring shield has an outside diameter of l consists of a 405-stainless-steel form enclosing that has magnetite iron ore as part of the aggre- is 6 ft high. A gas seal between the ring shield eactor building floor is provided by a low-melting- ure alloy poured into a trough.

tatable plug shield is constructed in much the nner as the ring shield except that casings extend through the shield to provide access to the core nts and additional thermal shielding is suspended bottom. Two large casings are 40 in. in diameter 20-in. In addition there are 81 small casings. in plugs that can be removed and reinserted by ling casks.

rge plugs are also concrete-filled stainless steel l are located so that they can be positioned over e reflector or moderator cans by rotating the plug The 75-ton plug shield is raised and rotated by a crane after the alloy has been melted in the tongue-and-groove seal which is located between the ring and plug shields. Three built-in rollers serve to center the plug shield as it is being rotated.

Reactor Core Structure

The SRE core structure consists entirely of a closely-packed array of hexagonal cans that contain the graphite moderator/reflector. These cans are supported and located on a stainless-steel grid plate near the bottom of the core tank. There are 43 moderator and 76 reflector cans in the core tank, each 11 in. across the flats and 10 ft high.

The moderator cans are fabricated of 0.035-in. zirconium sheet with bottom and top heads 0.1 in. thick. A stainless-steel support pedestal is bolted to the bottom head, and a stainless-steel spacer plate and lifting fixture is bolted to the top head. To control the pressure inside the can, a vent tube communicates to the helium above the free surface of the sodium. A zirconium tube 2.8 in. in diameter forms a channel in the center of the moderator can into which the fuel element is placed during loading.

The reflector cans are similar to the moderator cans except that they are stainless steel and do not have center channels.

Control and Safety Rods

The control and safety rods for the SRE operate in thimbles that project down into the reactor core from the top shield. In the original design, an attempt was made to keep the top surface of the shield as clear as possible of all control and safety rods to simplify the refueling operation. This requirement led to a rod design in which the mechanism that raises and lowers the poison column is a long screw on which rides a ball-bearing nut that supports the poison column. The screw is contained in a thimble but is located in a region below the thermal shield and in the upper sodium plenum. The screw and ball nut must, therefore, operate at temperatures higher than 960° F with the only possible lubrication being inorganic materials such as molybdenum disulfide.

Material failures at these temperatures forced a redesign of the control rod. In this second design, the entire mechanism is located in the thimble above the top shield where it operates at ambient temperatures. Two rods of the first design are being used for shim rods at the present. Two more rods of the second design are used as regulating rods.

The four safety rods are similar to the regulating rods in materials and configuration, but include a latch mechanism for dropping the rod and a snubber for deceleration. The latch mechanism consists of a chain drive attached to a magnet that holds the tube supporting the B-Ni rings.

Fuel Elements

The standard SRE fuel element consists of a cluster of seven rods suspended on a hanger that is attached to one of the small plugs in the top shield. Each rod consists of a 6-ft-high column of 6-in.-high slugs of alpha-rolled, beta-heat-treated, unalloyed uranium contained in a thin-walled 304-stainless-steel jacket tube with NaK for thermal bonding. The slugs are 0.750 in. in diameter, the NaK annulus 0.01 in. in average thickness, and the jacket tube 0.01 in. in wall thickness. The tube is closed at each end by a welded stainless steel plug. The NaK bonding alloy extends a few inches above the slug column to a free surface above which is helium at atmospheric pressure (when at room temperature). This space serves as a volume for the thermal expansion of the NaK and for the accumulation of fission gases.

To prevent the rods from touching each other or the coolant channel tube, the six outside rods are spirally wrapped with an 0.092-in.-dia stainless-steel wire with a pitch of about 10 in. Alternate rods are wrapped in counter directions to provide maximum mixing of the coolant. To determine the temperature of the exit sodium from any channel, a thermocouple runs down the inside of the hanger rods with its junction located just above the fixture at the top of the cluster. The fixture at the bottom of the cluster permits the expansion of the individual rods and includes an orifice plate and a locating guide.

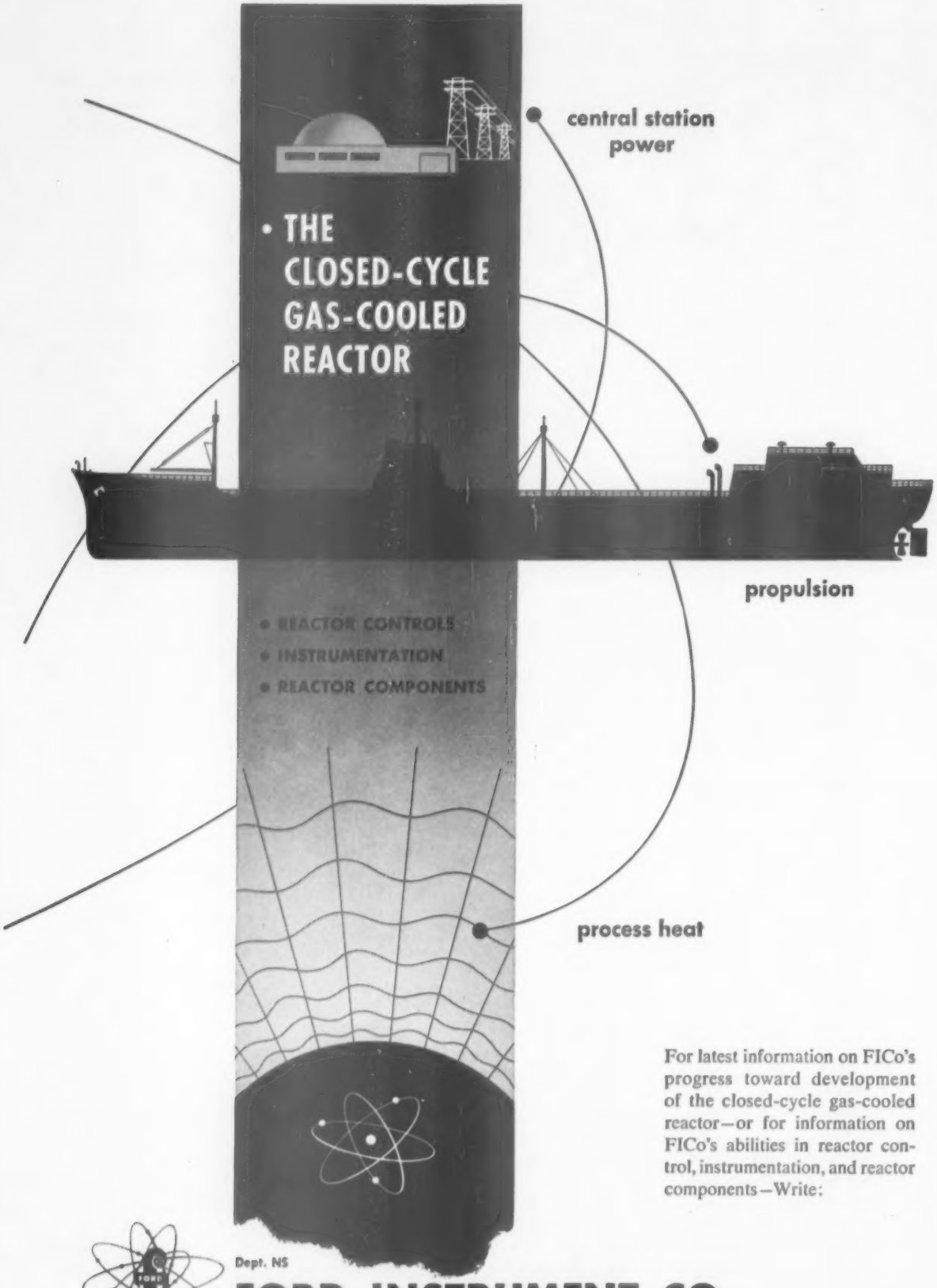
Sodium System

Although there are some problems concerned with the use of sodium as a coolant, solutions to these problems have been relatively simple. Because sodium has a fairly high melting point (208° F) and hence could solidify in a cold loop, heaters have been placed on all piping and tanks. Because sodium reacts chemically with oxygen and water, the large underground concrete vaults containing the primary sodium loops are filled with nitrogen, and helium is used as a blanket gas within the loops themselves. In addition, the water and steam in the secondary heat exchanger is separated from the secondary sodium by mercury in the annulus between the double-walled tubes.

Perhaps the outstanding advantage of the sodium-graphite reactor system is high thermal efficiency. It can produce steam at conditions approaching that available from modern conventional plants. Plans for the present SRE core call for reactor coolant temperatures approaching 1,000° F with steam being generated at 825° F and 600 psi.

Not only does SRE operate at high temperatures but it does so at low pressures. This is possible because molten sodium does not boil at atmospheric pressure until it reaches 1,614° F. This low-pressure operation means lower building costs and increased safety. The SRE does not require a containment shell or high-pressure vessels and piping. The reactor and the portion of the heat-transfer system that is radioactive operate at low pressure and are completely contained in underground concrete vaults. The materials used in the core and the heat-transfer system are compatible with one another, and there is no way in which energy within the reactor can result in the release of radioactivity in any significant quantity.

True, a special air-tight reactor, building is provided to contain any radioactivity release; however, the building is not a containment structure in the sense in which this phrase applies to other reactor types. The inherent characteristics of the sodium-graphite reactor and the individual containment provided by the fuel jackets, the stainless-steel heat-transfer system and the underground concrete vaults are adequate safety precautions.



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SRE GOES CRITICAL. Chauncey Starr (center), vice president and general manager of Atomics International Division of North American Aviation, Inc., and Sidney Siegel, director of technical operations, check recorder for log n count rate during initial criticality run

Starting Up SRE

Early low-power operation of the sodium reactor experiment has gone well. Lessons learned from planned experiments and a fuel-loading incident are being applied

By FRANK E. FARIS
Atomics International, Canoga Park, California

ONE OF THE PROMISING approaches to economic nuclear power lies in the development of the sodium-cooled reactor concept. The present embodiment of this concept is the Sodium Reactor Experiment (SRE), which has been designed and constructed by Atomics International as part of the AEC's five-year reactor development program.

Operations to date (to be described later in this article) have provided these valuable lessons:

- Observations during the power runs showed that the addition of flow-programming devices in the sodium systems and a pump to reduce leakage between the plenum feeding the fuel channels and that feeding the spaces between moderator cans would be desirable. These modifications will significantly reduce thermal stresses during operation at high power levels.
- It has been relatively easy to maintain the oxygen content of the sodium at 10 ppm with the cold traps under normal conditions.
- Fuel-handling procedures have been improved after study of a mishap that led to a damaged fuel element.
- Maintenance operations associated with the removal, cleaning, and replacement of the disposable cold trap for sodium oxides will be simplified by attaching the trap with flanges, rather than welding it in place.
- The rather slow approach to full power, which is required for the SRE because of its experimental nature and the fact that it has a large temperature difference across the core (460° F at full power) should not be required in the case of future plants designed with information available from the operation and the testing of SRE.

Precritical Tests

A dry subcritical fuel loading was performed, after all preliminary instrument and mechanical tests were completed, to obtain an extrapolated value of the critical mass without the primary coolant sodium. This experiment was performed by loading fuel elements from the center outward until the reactor was within two fuel elements of being critical. The predicted dry critical mass at room temperature from this experiment is approximately 22 fuel elements.

Upon completion of tests with the dry system the coolant system was preheated preparatory to filling with sodium. A number of cases of uneven heating were discovered during this period, particularly on some of the large sodium tanks. These situations were easily rectified by changes in the heater circuits and the addition of insulation.

Sodium was melted and transferred from the storage tanks to the reactor and the loops after they had been preheated to 350° F. Sodium was then circulated through the system to check out pumps and other components. Operational tests of the sodium systems were performed

with particular attention to the behavior of the process instrumentation and the adequacy of the cold traps for the reduction of the oxygen content of the sodium. It was found the cold traps maintained the oxygen content at 10 ppm. The hot traps have not yet been tested.

Critical Experiment

The wet-critical experiments were performed with the sodium in the core maintained at 350° F. Fuel elements were again loaded from the center outward with measurements for the inverse-counting-rate curve and estimates of control- and safety-rod worths being made after each element was loaded. SRE achieved criticality on April 25, 1957, with a wet-critical loading of between 32 and 33 fuel elements. At criticality there were 33 fuel elements in the core; three of the control rods were completely withdrawn, and the fourth was 5 ft above its lowest position.

Low-Power Physics Tests

A series of nuclear measurements at low power levels were then made that included control- and safety-rod calibrations, flux mapping, determinations of the worth of fuel elements in various positions, and a preliminary determination of the steady-state temperature coefficient of reactivity.

The worth of each control and safety rod was found to be about 1.2%. The preliminary temperature-coefficient measurements showed that the steady-state coefficient is positive ($\sim 3 \times 10^{-5}$/° F) in the low-temperature (400° F) range, a result that had been predicted by earlier calculations. The data indicate that it will become zero or negative above ~700° F, which is near the average core temperature during full-power operation.

It should be pointed out that the prompt coefficient, which is the coefficient of importance for control and safety, is negative. This fact was dramatically demonstrated by the self-regulating ability of the reactor during the first power runs when the reactor was operated for a time without the automatic neutron-flux controller. The only operator action required to maintain a constant power level was an occasional shim adjustment to compensate for the buildup of xenon and samarium.

Initial Power Runs

Before beginning the first power run, the over-all control and safety systems were checked out with the primary coolant system being maintained at 670° F. Then the fuel elements required for power operation were added to bring the total number of elements in the core to 36.

On July 12, 1957 the first power run was made, demonstrating the ability of the SRE to generate heat for conversion into electrical energy. During the first tests, with operation on an intermittent basis over a 3-day period, the turbine was on the line for 46.5 hr and produced a total of 59,800 kwh of electricity. The run was started at a power level of 2–3 Mw(th) with a 50° F temperature gradient (ΔT) across the core. This was followed by a gradual increase to 6.5 Mw(th) and a ΔT of 130° F. During the latter portion of the run, a maximum output of 1.7 Mw of electricity was reached. This is approximately one-third of the design power output. A subsequent power run, during which 18,400 kwh of electricity were successfully generated, was made on July 25 and 26, 1957.

The power runs provided the occasion for other closely related tests, such as the following:

Shielding. During both of the power runs, measurements of shielding effectiveness were made. The shield blocks above the galleries and the rotatable plug shield were found to be quite adequate. In fact, some measurements directly over intentionally unfilled gaps between gallery shielding blocks indicate that streaming through gaps is actually less than expected.

Flow programming. Systems analysis studies when the reactor was being designed had indicated that it might be desirable to program the flow decay to match the power decay subsequent to a "scram" so as to minimize thermal stresses, particularly in the moderator cans and in the core tank. Because of the many assumptions required for the calculations, it was decided to await actual reactor tests prior to a decision on the addition of programming devices. Several planned "scrams" were initiated during the power runs so as to measure the temperature and flow as a function of time following the dropping of the safety rods. The results of the tests indicated the desirability of the addition of electromagnetic eddy-current brakes to slow down the coolant at a rate that matches the decay in power during a shutdown.

Moderator cooling. During the power run the moderator can cooling system was checked. It was found that there is some leakage between the two lower plenum chambers with the result that the moderator can cooling is greater than anticipated. This situation is being corrected by the addition of an electromagnetic pump, which will pump sodium from the plenum feeding the spaces between the moderator cans back into the main primary sodium system.

Normal Operation

It is interesting, in the context of the foregoing discussion of early operation of SRE, to describe the procedures that are normally used to bring the reactor and steam system to full power (see figure on p. 39).

To begin with, the reactor safety system and interlocks are tested before starting a power run. The sodium is then heated by the electrical heating system to 350° F, and the coolant flow rate is maintained at 100% in all sodium loops, a procedure designed to provide maximum safety during the initial phases of startup. With the neutron source providing a constant supply of neutrons the withdrawal of all four safety rods is followed by the withdrawal of the control rods while the operator observes the nuclear instrumentation.

When reactor power (neutron level) is seen to increase steadily without further control-rod withdrawal, the reactor has become critical. Power is then increased to approximately 1% of full power, and the electrical heaters for the sodium system are turned off. Power is gradually increased at a rate which will give an increase in sodium temperature of 30° F/hr. This rate of increase is probably conservative but will be adjusted as more is known about the system. When the sodium in the secondary loop reaches a temperature of 440° F, feedwater flow to the steam generator is started and is adjusted to hold the steam-generator-outlet sodium temperature at 440° F.

Power is then increased until 15% of full power is

reached. At this point the power level is stabilized, and the reactor is placed on automatic neutron level control. Sodium flow is then decreased at a rate that will give an increase in reactor outlet temperature of 30° F/hr. The inlet temperature is, meanwhile, held constant by the adjustment of the boiler feedwater flow. When the sodium flow has been reduced to 15%, the full 460° F gradient is established across the reactor, with the sodium inlet temperature at 500° F and the outlet temperature at 960° F. The reactor is then returned to manual control.

The turbogenerator in the steam plant is started at this point and allowed to attain its correct speed. When the generator is synchronized and on the line, the reactor control room is advised that the steam plant is ready to receive additional power. Sodium flow and reactor power are then increased together in such a way that the reactor outlet temperature is held constant. This increase is continued until the reactor is at full power with 100% sodium flow. At this point the reactor is again placed on automatic control. Boiler feedwater will have been placed on automatic control whenever the feedwater flow was sufficient to permit automatic regulation.

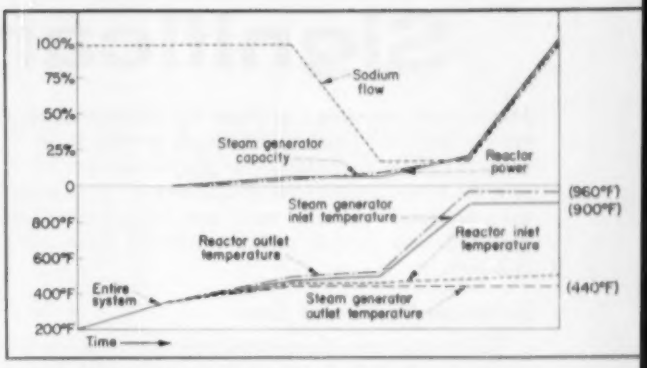
Maintenance Experience

The two most important maintenance operations performed thus far are the removal of a damaged fuel element from the core and the replacement of the disposable cold-trap cartridge. In both instances the operation provided valuable experience and generated confidence in the maintenance techniques adopted for SRE.

Fuel element. The damaged fuel element was discovered when difficulty was encountered in trying to remove it with the fuel-handling cask subsequent to the second power run. When the shielding plug was finally freed, it was found that the hanger rod had parted and that the 7-rod cluster had remained in the core. The sodium level was lowered, and examination showed that the cluster was protruding approximately 18 in. above the top of the moderator can and that the protruding portion was bent at approximately 30° to the vertical. The cluster and other broken pieces of the assembly were removed in a single day with grappling tools devised for the purpose.

The investigation of operating records indicated that the element, which had been loaded between the wet-critical experiment and the first power run, had never occupied its proper position in the core. This discovery explained a small decrease in reactivity that had been observed prior to the power runs. The records showed that the fuel element in question was loaded at a time when the sodium level had been lowered to the tops of the moderator cans. Calculations indicate that under these conditions freezing of sodium can occur when a cool element enters the fuel channel. This freezing of sodium as the element was lowered undoubtedly explains why the element never completely entered the core. Operating procedures have now been modified so that fuel and other core elements are never loaded without the presence of the sodium pool above the core, and a soaking period in the pool is now specified to insure that the elements will be hot enough to prevent freezing before they are placed in the core.

Cold trap. As a result of the initial cleaning of the sodium and the cleaning to remove oxygen introduced into the system during the removal of the damaged fuel element, it was found that the disposable cold trap for the main primary system had become inoperative. The trap was cut from the system and steam-cleaned without incident. A new cartridge was then installed, and the trap was replaced. An improved trap, in which the volume available for effective oxide deposition is increased, has been designed. Flanges are incorporated in the new design for easy removal when the trap has become filled with oxide.



SRE STARTUP PROGRAM. Normal operation begins with 100% sodium flow and entire system at ~350° F. Full power is achieved some hours later with reactor inlet temperature of 500° F, outlet of 960° F. Primary sodium system operates at 3 psi, secondary system at 40 psi, and steam is generated at 600 psi and 825° F

Future Operating Plans

Further power runs at relatively low levels are in progress at the present writing (November, 1957). Subsequent to these runs, the low-power physics experiments will be completed; plant modifications suggested by the first three power runs will be carried out; and the hot traps will be tested and placed in operation.

Full-power operation is planned for early next year. As soon as full-power operation has been achieved, heat balances, reactivity changes, graphite outgassing and the like will be measured. Particular emphasis will be placed on achieving high integrated power to obtain information on the behavior of the various fuel elements being tested in the core.

Fuel-Element Development

The SRE fuel-element program is of prime importance because it holds the promise of higher-temperature operation with consequent improvement in thermodynamic efficiency. Several experimental fuel elements have already been fabricated for testing. After completion of the initial power runs, it is expected that from 20–25% of the fuel elements will be experimental. Among the elements to be tested will be ones containing centrifugally cast uranium, centrifugally cast uranium-zirconium and uranium-molybdenum alloys, uranium-molybdenum alloys fabricated by power metallurgy, extruded thorium-uranium alloys, and uranium fabricated by powder metallurgy in the form of large hollow cylinders rather than in the form of the standard seven-rod clusters. A program is also underway involving various ceramic fuels that seem to hold particular promise for long burnup at high temperatures.

Significance of SRE

MUCH MORE operating experience will be required before one can judge the ultimate significance of SRE as an embodiment of the sodium-graphite reactor concept. However the design, construction and limited operation to date have produced problems, some now solved, others still unsolved, that provide useful lessons for other reactor designers.

Zirconium Moderator Cans

The development of zirconium moderator cans has been one of the principle design problems in the SRE project. Tests on the behavior of zirconium in circulating sodium showed that Zr is susceptible to oxidation even at very low levels of oxide impurities in sodium. Because of this great affinity of zirconium for oxygen, a surface layer of oxide develops. This layer of oxide is brittle and cracks easily under stress. These surface cracks can then lead to accelerated fatigue failure of the zirconium during alternating stress cycles.

This problem has largely been solved in the present SRE core by a can design that strengthens the regions of high stress (NU, June '57, 84) and by the inclusion in the primary loop of an oxide-trapping hot trap in addition to the usual cold traps. The hot trap reduces the oxygen content of the sodium to a level substantially lower than the 10 ppm attainable with cold traps alone.

Fuel Element Experience

The metallic uranium slugs chosen for the SRE fuel element seem to be under a cloud at the present time. The results of recent in-pile tests of SRE fuel elements in the MTR have been unfavorable. The test results were not too conclusive because the neutron flux may have been higher than anticipated. This would have lead to test temperatures far in excess of reactor temperatures. Thus, the MTR tests do not settle the question of the effects of high temperatures on the present fuel element design. Still, they cause doubt and provide an argument for caution in bringing the SRE up to full power.

Clearly it is not possible to predict how long the current fuel elements will last under full-power conditions. Original estimates of 2,500 Mwd/ton are generally regarded as optimistic. Surer answers can come only when the reactor goes to full power early next year.

The in-pile tests at MTR did furnish one positive conclusion: thorium-uranium combinations seem superior to straight uranium. Coupled with AI's plans for oxide fuel elements, this provides lots of possibilities for the over-all SRE fuel-element development program (see p. 39).

It is also planned to replace the present core with a completely new second core, including a different configuration of the graphite moderator, possibly a different method of canning the graphite itself, and a considerably different fuel element design.

This second core for SRE is an attempt to achieve sodium temperatures approaching 1,200° F (vs 960° F). The tanks and piping in the present reactor and sodium system have been designed for these higher temperatures. It is estimated that $4.7 million will be required for these modifications and for a sodium flushing and solid-waste storage system not now included in the SRE plant.

Reactor Physics

From a reactor physics point of view, a number of important measurements have already been made, particularly those relating to critical mass, worth of fuel elements in various positions, control- and safety-rod worths, neutron-flux distributions and shielding effectiveness. A preliminary measurement of the temperature coefficient has been made, and, as predicted by calculation, the *steady-state* coefficient has been found to be positive in the low-temperature range ($\sim 3 \times 10^{-5}$/° F at 400° F). The data indicate that the coefficient will decrease with increasing temperature and become zero or negative ~700° F. The *prompt* temperature coefficient, which is the important one from a safety and control standpoint, is negative. With a negative prompt coefficient available to provide stability and safety, the existence of a positive steady-state coefficient is an advantage in that a smaller loading of excess fuel is required for operation.

Continued experiments on the physics of the reactor will be carried out with special emphasis on the worth of the experimental fuel elements, more careful measurements of the steady-state temperature coefficient, measurements of the prompt temperature coefficient, and the determination of the effect of long burnup on reactivity. The results of these studies will be of importance in connection with the core design of full scale sodium-graphite reactors.

At present the steam plant is subservient to the reactor. After the experimental program on the reactor is well advanced, it is planned to change the control scheme so that the reactor will follow steam plant demands. The technical knowledge being gained in the operation of the steam plant will be available to other utility companies.

What Has SRE Cost?

The cost of the SRE project through fiscal year 1957 totals $16,733,000, of which $2,850,000 was borne by NAA. A breakdown of this total follows.

	Cost of SRE by Fiscal Years (thousands of dollars)				
	1954	1955	1956	1957	Total
Research & development	1,311	1,566	2,254	3,814	8,945
Reactor fabrication	133	858	4,280	2,008	7,279
Fuel fabrication	0	135	80	113	328
Test operation	0	0	1	180	181
Total	1,444	2,559	6,615	6,615	16,733

An additional $8,271,000 will be spent during the current fiscal year for research and development and operating expenses. This brings the total cost for the term of the original contract to $25,004,000. This is 150% more than the $10 million that was estimated as the total cost when the contract was awarded on June 25, 1954.

M. W. Kellogg Co. radioisotope laboratory in Houston, Texas, scene of a radiation incident March 13, 1957

OFFICIAL INQUIRY REPORT ON NUCLEAR ACCIDENTS

NUCLEONICS

1. Kellogg Radiation Incident

2. Windscale Reactor Accident

Britain's Windscale Reactor No. 1, foreground, was put out of commission by accident on October 10, 1957

In the past decade, the world-wide nuclear energy industry has amassed a truly amazing safety record. Remarkably few accidents have taken place in the handling of nuclear materials and in the operation of nuclear facilities.

Although public interest in matters of nuclear safety has always been high, this interest is increasing considerably as more and more individuals and organizations become involved in the nuclear field. It is thus important not only for the welfare of the general public but also for the well-being of the nuclear industry that the public be enlightened on such matters.

With this in mind, NUCLEONICS plans to publish the full texts (space permitting) of official reports on nuclear accidents as quickly as such reports become available. Speed is vital because delays in issuance of the facts permit false information to be circulated.

The first such reports are published in this issue. On p. 43, there appears the full text of the Atomic Energy Commission report of the incident that took place at the M. W. Kellogg plant in Houston, Texas, on March 13, 1957. Starting on p. 44 there appear the portions of the report relating to the causes of the accident that took place at the United Kingdom production reactor facility at Windscale on October 10, 1957.

Turn page for official texts of accident reports ➡

M. W. Kellogg Co. Radiation Incident on March 13, 1957

This is a statement by the General Manager of the Atomic Energy Commission concerning the background and facts relating to an incident involving the uncontrolled release of radioactive materials at the South Houston plant of the Nuclear Products Division of the M. W. Kellogg Company, in March 1957. The information in this statement was gathered by direct observation by the AEC staff, interviews of persons involved in the incident, and review of the records of the licensee, including information provided to the licensee by the company that supervised the decontamination of all facilities, areas and objects contaminated as a result of the incident.

On April 19, 1957, the M. W. Kellogg Company, 711 Third Avenue, New York 17, New York, notified the Atomic Energy Commission that an incident had occurred at its South Houston, Texas, facility involving the uncontrolled release of radioactive iridium 192 in the company's laboratory.

At the time of the incident, March 13, 1957, the M. W. Kellogg Company held five valid AEC licenses as follows:

(1) Auth. No. 37437 for 1,500 curies of cesium. Expiration date, March 31, 1957.

(2) Auth. No. 37438 for 6,500 curies of iridium as "metallic" sealed sources, irradiated at MTR. Expiration date March 31, 1957.

(3) Auth. No. 38176 for 1,000 curies of iridium-192 as sealed sources. Expiration date March 31, 1957.

(4) Auth. No. 34354, A, B, C, D, E. Expiration date December 31, 1956. Application for renewal of this authorization was received by Isotopes Extension, November 1956.

(5) License No. 31-246-1 for 500 curies of cobalt-60. Expiration date April 30, 1957.

License No. 31-246-2 was issued to the company on April 2, 1957, subsequent to the incident but prior to the incident being reported to the Atomic Energy Commission. This license superseded licenses referred to in (1), (2), (3), and (4) above.

The authorizations were issued to M. W. Kellogg Company, Nuclear Products Division, and provided for the encapsulation of radioisotope sources for sale in gamma ray exposure cameras to persons holding valid AEC licenses. License No. 31-246-1, for 500 curies of cobalt-60 as sealed sources, was issued to M. W. Kellogg Co., Construction Department, for nondestructive-testing purposes. The licensee had previously been issued several other authorizations which had expired and are not pertinent to the incident.

The licensee was, at the time of the March 13, 1957, incident, subject to 10 CFR Part 20, "Standards for Protection Against Radiation," which became effective on February 28, 1957.

The M. W. Kellogg Company had prepared two manuals dealing with training and administrative operating instructions for the company. The Kel-Ray Training Manual was dated December 5, 1955. The company had also prepared a "Safety Manual" dated November 15, 1955, which was available to employees. This Manual contained specific sections on safety policy, safety practices, job inspections, and reports to the home office for both field and laboratory work.

On or about March 13, 1957, two M. W. Kellogg Company employees, one in protective clothing and respirator and one in street clothes, were engaged in opening a sealed aluminum can containing 10 radioactive iridium sources consisting of iridium-aluminum pressed pellets $\frac{1}{8}$ in. × $\frac{1}{8}$ in. right cylinders. Each pellet consisted of approximately 35 curies of iridium-192. To permit the removal of the plug, containing the pellets, from the can, the end of the can was cut with a jeweler's lathe in a sealed plexiglass box which in turn was in a "hot cell" consisting of 33-in.-thick concrete walls, a shielded window, and an open top. Remote control slave manipulators were used by the operators. This operation was completed in the morning.

The can was allowed to remain in the sealed plexiglass box for a period of time to permit any contamination to settle in the box. After lunch an employee, again dressed in protective clothing consisting of white pants, shirt, shoes and respirator, opened the sealed box. The second employee, who was present that morning, was again present in street clothes. The can was removed from the box and placed on a paper mat on the floor of the hot cell, and when the plug was removed from the can it was observed that portions of the first two pellets were in particle form. Some of the particles fell onto the paper mat. During this operation the air monitor device in the laboratory indicated the presence of airborne radioactivity. Subsequently, the employee in street clothes left the laboratory. The employee wearing protective clothing remained in the laboratory. The exact extent or timing of the work he then performed is not known but it did involve the placing of the 8 intact and 2 partial pellets, together with the particles of iridium-192, in the storage pit at the bottom of the hot cell. Before this employee left the hot cell he placed the storage pit lid in place. He removed his protective clothing, took a shower, and dressed in his street clothes before leaving the Kellogg premises.

Between the afternoon of March 13, 1957, following the removal of the pellets from the plug, and on or about April 10, 1957, the following events occurred:

1. The 8 intact and 2 partial pellets were encapsulated and placed in a storage well.

2. The air monitor in the laboratory was placed back in service.

3. The jeweler's lathe and paper mat were placed in a waste storage pit.

4. The hot cell was washed with soap, water and sponges, which lowered the radiation level from approximately 2000 mr/hr to about 20 mr/hr.

5. The jeweler's lathe was removed from storage pit and sprayed with soft spray. After completion of this procedure the lathe showed a reading of 5000 mr/hr. It was dismantled, placed in a 5 gallon can of kerosene and lead bricks were placed around it.

On or about April 11, 1957, an employee entered the laboratory, removed a cesium-137 source from the storage well and loaded it into a projector. At that time the Manager, Nuclear Products Division, noted a high radiation reading outside of the iron doors at the end of the hot cell.

Subsequent to the Manager's observation of contamination in the vicinity of the hot cell, he made further inquiries concerning the spread of contamination. During the following week he determined that the laboratory area was contaminated, that employees' street clothes and shoes had been contaminated as a result of the incident and that some radioactive material had been carried into employees' homes and automobiles. On April 19, 1957, he notified the AEC of his findings.

The last entry into the laboratory was on or about April 22, 1957, when the Manager, Nuclear Products Division, entered to procure some equipment. It was then closed down until decontamination work was begun. AEC investigation was initiated April 22, 1957, and on May 2, 1957, the AEC issued a temporary suspension order on License Nos. 31-246-1 and 31-246-2, which suspended not only the Houston laboratory operation but all AEC-licensed activities of the Kellogg Company.

On May 17, 1957, the Commission issued License No. 31-246-3 to the company, which had the effect of partially lifting the May 2

suspension order. That license authorized the company's Construction Department to possess and use certain radioisotopes as sealed sources in radiographic testing of welds and castings. This work was previously carried out by the company under License No. 31-246-1. The activities of the Construction Department are of a different nature and conducted in different facilities from the activities of the Nuclear Products Division at South Houston, Texas.

On or about April 24, 1957, the Kellogg Company employed Tracerlab, Inc., of Boston, Massachusetts, to survey and decontaminate all areas involved as a result of the incident. This procedure was initiated on or about the morning of April 25, 1957, and of 53 automobiles checked, 7 showed some evidence of contamination. Of 19 homes checked, 8 gave evidence of contamination. Several articles of clothing were also found to be contaminated. The survey and decontamination procedure was completed during May 1957, except for one automobile and one trailer which were isolated in April 1957 and completed in August 1957.

All objects and areas outside the Kellogg plant suspected of contamination have been monitored by radiation detection instruments and decontaminated where contamination was found.

The Kellogg Company arranged for medical examination of 20 persons, 19 of them employees or members of their families and one a neighbor of an employee. These persons were examined under the direction of Dr. Herbert C. Allen, Internist and Director of the Radioisotope Department of the Methodist Hospital, Houston, Texas. Except for a minor radiation burn noted on each of the two laboratory employees who were present at the time of incident, medical examinations were negative. Complaints of nausea, vomiting and loss of weight by one of the employees who received one of these minor burns have been evaluated by the examining physician as having no pathologic organic relation to his exposure to radiation. When last examined by Dr. Allen the burns on both employees were healing without complication.

Records, maintained by the licensee, of radiation exposures received by employees while working in the laboratory area show that neither of the two employees present in the laboratory at the time of the incident received a radiation exposure, during the week the incident occurred, in excess of permissible values established in the AEC "Standards for Protection Against Radiation."

The records indicate that during the week ending on or about April 12, 1957, about four weeks after the incident occurred, one of the employees received an exposure of 1,720 millirems which is about twice the exposure permissible under the regulations for any period of seven consecutive days. This occurred during the clean-up operations in the laboratory.

This one employee's cumulative exposure record for 13 consecutive weeks, including the week of the incident and the April 12 weekly exposure, was 3,875 millirems. This exposure exceeds the permissible quarterly exposure, under these conditions, by about 25% of that permitted by regulation. There were no other overexposures to employees recorded during that period.

All areas within the plant site have been either completely decontaminated or established as restricted areas until they can be completely decontaminated.

Accident at Windscale No. 1 Pile on October 10, 1957

Report of the Committee of Inquiry

The cause of the accident and the measures taken to deal with it

EDITOR'S NOTE: *After the Windscale accident (NU, Nov. '57, p. 130) the United Kingdom Atomic Energy Authority appointed a four-man Committee of Inquiry to investigate causes and measures taken to deal with the accident and its consequences. Members were Sir William Penney, Director of the U. K. AEA Atomic Weapons Research Establishment; Dr. B. F. J. Schonland, Deputy Director of the Atomic Energy Research Establishment; J. M. Kay, Professor of Nuclear Engineer at Imperial College of Science and Technology; Jack Diamond, Professor of Mechanical Engineering at Manchester University. In addition a special committee established by the Medical Research Council investigated health and safety aspects.*

Reports of these investigators were published on November 6 in a 26-page document consisting of a memorandum by the Prime Minister and 6 "annexes." The first annex is an abbreviated version of the report of the cause. The others deal respectively with measures to deal with consequences; health and safety aspects; a memorandum by Plowden; a note by the U. K. AEA on Calder Hall and civil-power reactors; and the assignments and memberships of the committees. The first and fifth annexes are reprinted here.

Copies of the complete Windscale report can be obtained by sending $0.28 to British Information Services, 45 Rockefeller Plaza, New York 20, N. Y.

Summary

1. The accident occurred during a controlled release of stored Wigner energy from the graphite of the pile.

2. The immediate cause of the accident was the second nuclear heating. This was applied before it was necessary and the nuclear heating was put in at too rapid a rate. The second nuclear heating was applied because it was thought that the Wigner release was dying away and that parts of the graphite structure were therefore going to escape being annealed. The instrumentation of the pile was not sufficient in quantity or in distribution throughout the pile to enable a reliable judgment to be made.

3. The second nuclear heating led to the accident by causing the failure of one or more cartridges whose contents then oxidised slowly, eventually leading to the fire. By far the most likely possibility is that the second nuclear heating caused the failure of one or more uranium fuel cartridges. A second possibility, which cannot be entirely eliminated, is that it was a lithium-magnesium cartridge which failed. The second nuclear heating may have released pockets of Wigner energy at a time when the general level of temperatures throughout the pile was high.

4. During the afternoon of Thursday 10th October, the operating staff of the pile thought they had a bad burst of a fuel cartridge. At 16.30, visual inspection through a plug hole in the pile wall revealed glowing fuel cartridges. Various actions were taken and by about midnight the major decision had been made that if other measures failed, water was to be used to reduce the fire. Shortly after midnight, the Chief Constable of Cumberland was warned of the possibility of an emergency, and men in the factory were warned of an emergency with instructions to stay indoors and wear face masks. Water was actually turned on at 08.55 on Friday 11th October. Water was turned off at 15.10 on Saturday 12th October, by which time the pile was cold.

5. The Committee of Inquiry advised the Atomic Energy Authority that in their opinion the steps taken to deal with the accident once it had been discovered were prompt and efficient and displayed great devotion to duty on the part of all concerned.

The Cause of the Accident

6. The phenomenon of the storage of energy in graphite due to neutron bombardment (the so-called Wigner energy) has

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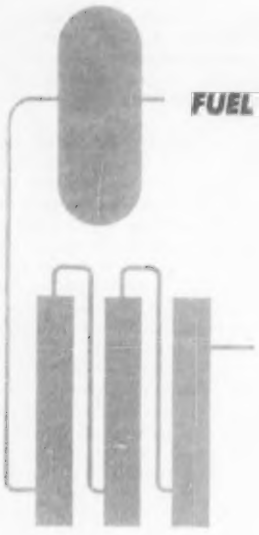

FUEL REPROCESSING SERIES—2

Low-Decontamination and Volatility Methods

By EDWARD L. ANDERSON, Jr.
Division of Reactor Development, U. S. Atomic Energy Commission, Washington, D. C.

SEVERAL FUEL-REPROCESSING schemes under development today show promise for the long-term nuclear-power picture. They offer advantages over the conventional solvent-extraction processes* in plant size, waste handling, range of fuel materials handled and, probably, cost. These schemes are grouped here as volatility and low-decontamination processes.

Volatility processes (Fig. 1), like solvent-extraction processes, are able to achieve high decontamination—removal of essentially all radioactive contaminants (fission and corrosion products) to permit manual refabrication. The basis of these processes lies in the differences in volatility of the uranium, plutonium and fission-product fluorides. Two approaches are being studied (*1*)—low-temperature (120° C) fluorination with BrF_3 and high-temperature (600–700° C) fluorination in fused salt with HF and then F_2.

Several other approaches to volatility processing have been considered informally. These include the Zircex-type headend (*2*) followed by a transformation of the UCl_3 product into UF_4. The fluorination techniques described in this article for the high- and low-temperature processes would be applicable. Furthermore, the diluting zirconium alloy would be separated from both the uranium and the fission products.

Volatility processing compares favorably with solvent extraction in six respects: (1) Decontamination is as good or better. (2) Process is chemically simpler. (3) Waste is easier to handle as a concentrated solid. (4) Wider range of certain fuel materials (zirconium-niobium) can be handled. (5) Cost advantages are anticipated. (6) Reagents are probably less susceptible to radiation damage. However, plutonium, particularly the hexafluoride, presents a handling problem because of its reactivity.

Low-decontamination processes have as their objective the purification of reactor fuels only to a level such that they are suitable for reactor recycling. The distinction between the high- and low-decontamination processes in terms of decontamination factors (DF) is rather sharp—high decontamination refers to DF's of the order of 10^6–10^8, whereas low decontamination refers to DF's of the order of 10–100 (perhaps for only certain fission products).

Aqueous and pyrometallurgical processes are being studied for low decontamination. Because the products of these systems are too radioactive for manual handling, the processes must be used for continuously processing liquid, homogeneous fuels or in conjunction with remote refabrication of solid fuels. Although the processing cost is high, this should be compensated for by a lower fuel inventory then that required by high-contamination processes with their long pre- or post-processing cooling periods. Low-decontamination wastes are also in a solid, compact form.

Aqueous reprocessing cycles are described here for the Fast Oxide Breeder (FOB) and Homogeneous Reactor Experiment (HRE-II). In both cases small low-decontamination plants are used for recovery of the valuable core material while blanket material is transported to central processing plants for more conventional extraction processing. Both core cycles use a solvent-extraction step after a preliminary leaching step for the FOB and a separation of solid poison material from the HRE-II fuel solution.

Pyrometallurgical processes (Fig. 2) use a molten-metal medium for oxidation-reduction reactions or for separation by volatilization, crystallization, or extraction. An important feature is the high radiation resistance of the reagents. Processing cycles are described here for the Experimental Breeder Reactor and liquid-metal-fueled reactors.

* The first part of this series, "Fuel Reprocessing by Aqueous Methods," appeared in NUCLEONICS, Oct. '57, p. 72. The third and fourth parts, appearing in future issues, will cover waste disposal and the economics of processing.

Volatility Processing

It was long recognized that a separation method based on UF_6 volatility (Fig. 3) would eliminate several of the steps that are part of current solvent-extraction techniques—(1) solvent-extraction cycles themselves, (2) the denitration of the uranyl nitrate product to UO_3, (3) the subsequent H_2 reduction to UO_2 and (4) hydrofluorination to UF_4, which is reduced to metal or fluorinated to UF_6 for enrichment by gaseous diffusion.

Plutonium separation from a volatility system is a severe problem. In the present advanced state of aqueous-separation methods for natural-uranium systems, the original proposal of a volatility process for uranium separation followed by an aqueous process for plutonium separation from the fluoride salt residue may no longer be competitive.

Several inherent advantages of volatility systems have, however, only recently become apparent, and an extensive development program is still underway. Interest in these systems centers around their potential for simplifying the problem of ultimate waste disposal as well as their advantages in coping with the problems presented by zirconium-type fuels.

The waste-disposal advantage arises from the fact that the process waste is already concentrated, either as a dry solid in the BrF_3 dissolution vessel or as a solution of fission-product fluorides in the molten-salt dissolution medium. One of the major efforts in treating wastes from aqueous processing streams is aimed at developing condensed phases to reduce the fission-product mobility. Salt residues of volatility processes may well accomplish this directly. In addition, the current aqueous zirconium process, as well as several alternates under investigation, requires the use of fluoride ion, the corrosive nature of which has been found to compound severely all the usual difficulties associated with interim and ultimate waste storage.

Additional interest in volatility processes has arisen from the realization that present aqueous methods, for zirconium fuels in particular, are of only limited applicability at this time. The only process for zirconium-alloyed fuels that has been fully developed is limited, for the moment at least, in the range of Zr-U ratios that it can handle. No such difficulties are anticipated in the high-temperature volatility processes, which offer excellent potential with respect to a wide variety of zirconium- and niobium-base fuels and may be satisfactory for other types.

The future of the high-decontamination volatility approach will depend, to a large extent, on continued success in solving the engineering problems.

The BrF_3 approach (Fig. 4), assuming the "all-volatility" plutonium process is developed, is limited by the same problem found in aqueous processing—only declad or chopped fuel elements can be dissolved directly. Separate decanning steps, probably aqueous, might be required to remove jackets of stainless steel, aluminum or zirconium. The fused-salt process, while probably applicable to most alloyed fuels, has been directed primarily toward the zirconium types until recently, and its versatility has yet to be demonstrated.

Both these processes would probably be suitable for ceramic fuels. However, neither appears useful in thorium processing (if thorium is to be recovered) inasmuch as no adequately volatile thorium fluoride is known.

With respect to their potential for short-cooled fuel reprocessing, the volatility processes compare favorably with aqueous techniques. Volatility reagents are, if anything, less susceptible to irradiation damage, and although the short-cooled fission product spectrum leads to larger quantities of the theoretically more volatile iodine and tellurium fission product fluorides, separation can still be satisfactory.

Low-Temperature Processes

In the low-temperature BrF_3 process, the over-all reaction is that of F_2 reacting with uranium metal to give UF_6. BrF_3 is used as an intermediate due to the difficulty of reacting uranium metal with fluorine gas directly. With BrF_3, which can be maintained as a liquid under pressures of about two atmospheres and temperatures of about 120° C, the reaction can be readily controlled. Reaction must not be permitted to occur in the vapor phase due to the difficulty of removing the heat of reaction rapidly enough to prevent ignition of uranium metal. The dissolution reaction produces, among other things, bromine and bromine monofluoride, which are in turn oxidized with F_2, reforming BrF_3, which

Two Promising Nonaqueous Processes

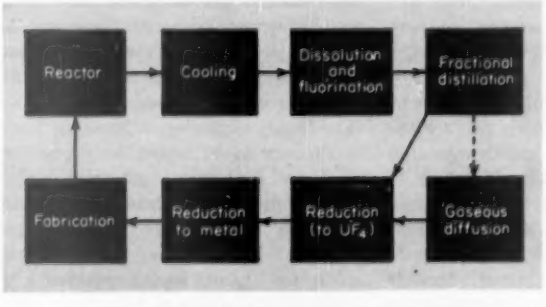

FIG. 1. Volatility processing, like solvent extraction, is designed for high decontamination—it removes sufficient radioactive contaminants to permit direct, conventional fabrication—but plant would be smaller and wastes more compact. (Diagram courtesy of Argonne National Laboratory)

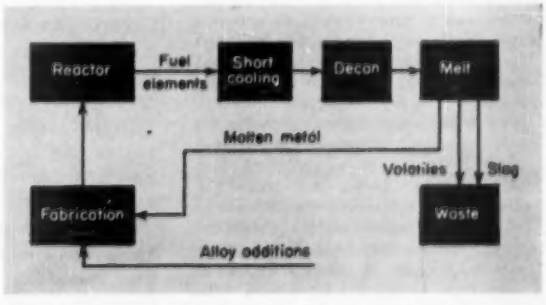

FIG. 2. Pyrometallurgical processing is one of the low-decontamination methods—enough of the radioactive poisons would be removed to meet reactor reactivity requirements. Advantages are small plant and compact wastes; disadvantage is costly remote refabrication. (Diagram courtesy of ANL)

is then recycled to the dissolver vessel.

Pilot-plant operations using fully irradiated material have yielded uranium products decontaminated to a degree at least as high as that obtained in conventional aqueous processing. However, bromine contamination in the product UF_6 is still slightly high.

Construction materials. Process development thus far has centered on problems of corrosion and design of valves, fittings, pumps, etc. Extensive corrosion studies have been made to find suitable construction materials.

Nickel and the high-nickel alloys, such as Monel, are generally satisfactory. The existing pilot plants have been constructed almost exclusively of such materials. In part, such a choice resulted from the original intention of contacting the dissolver with both inter-halogen and aqueous solutions when an aqueous recovery of residual plutonium was being considered.

Relative Electropositivities of Elements in Spent Fuel

Cs, Ba, Rb, Sr		Volatile
Pr, La, Pm, Ce, Nd, Y		Separations feasible using appropriate oxidation-reduction reactions
Pu, Zr, ThU	Mg, Nb	Separations feasible using appropriate oxidation-reduction reactions
Mo, Rh, Ru		Redox inapplicable without reacting bulk U as well

These data are useful for redox reactions used in pyrometallurgical processing. Each metal can replace oxidized form (e.g., chloride, fluoride, oxide) of metal below it; degree of replacement is determined by magnitude of difference in electropositivities. Volatile fission products include krypton and xenon. Degree to which alkali and alkaline-earth metals are volatilized depends on temperature and pressure of system; under certain conditions these elements may be oxidized rather than volatilized. In practice, degree of separation is severely limited by difficulty of maintaining fine control in pyrometallurgical operations. Separation by oxidation-reduction of elements below U, Th or Pu is impossible without also treating U, Th or Pu

Pu separation. Significant progress has been made on the "all-volatility" process in which plutonium is volatilized from the salt residue. It has been demonstrated in the laboratory that, at reasonable temperatures, PuF_4 can be fluorinated at satisfactory rates directly to the volatile PuF_6. Considerable additional effort will be required, however, to master the problems of maintaining and transporting the material under process conditions. Furthermore, little attention has yet been given to the actual distillation separation of PuF_6 from those few fission products that are significantly more volatile at the anticipated higher operating temperatures.

High-Temperature Process

The fused-salt-dissolution or high-temperature volatility process is of course quite similar to the low-temperature processes. An important area of difference is that two distinct fluorination steps are used, the first being the reaction with HF to produce the lower fluorides and the second, the treatment with F_2 to obtain the volatile UF_6.

While F_2 could perhaps be used in both steps in view of the liquid-salt heat-removal medium present, there is an undoubted safety advantage in the dual procedure. Furthermore, F_2 is more expensive than HF.

Construction materials. In this process, also, selection of satisfactory construction materials is of paramount importance. High-nickel alloys are, as expected, the immediate choice. Complications are involved in their use, however, because of the 600–700° C temperature of operation and the presence of the fused-salt medium. Ordinary "A"-type nickel is subject to precipitation of its contained carbon at the temperatures indicated; "L"nickel, since it has a lower carbon content, does not have this disadvantage. Both kinds of nickel are highly susceptible to intergranular attack from sulphur-bearing materials (*2*). These effects are additive and irreversible. Special precautions must therefore be taken to utilize only the very purest salts and HF in the process. Under these conditions "L" nickel can be used satisfactorily. Monel, Inconel, Hastelloy-B and copper have also been considered.

Graphite vessels offer considerable promise in the hydrofluorination step, especially when the heat is introduced to the center of the vessel and the vessel itself is cooled externally, thus forming a frozen salt layer, or liner, as a corrosion-resistant film on the inner walls of the vessel. Graphite's lack of impact strength is a disadvantage and it may be necessary to design methods for introduction of fuel elements to the dissolver that are more elaborate than the simple dropping methods currently in use.

Graphite is unfortunately not satisfactory as a container for the fluorination step in this process. Nickel appears promising, although, if graphite is used in the hydrofluorination, it will be necessary to transfer the fused salts to another vessel before the fluorination step itself. It has also been suggested that the dissolution or fluorinating vessel containing the fission products be discarded after several batches and used as the waste-disposal container. Under these conditions less expensive materials of construction might suffice although operation might be difficult.

No completely satisfactory solution to the dissolver-material problem has as yet been found. It may be possible to tolerate relatively high corrosion rates and the resulting rapid discarding of dissolution vessels in view of the relatively high value of the enriched uranium material that can be processed per dissolution batch.

Pilot-plant testing of high-temperature volatility processing is about to get underway. In all likelihood, both graphite containers and low-carbon-nickel containers will be tested, with both the distillation and the adsorption-desorption methods being tried with fully irradiated material.

Hazards

In addition to the usual difficulties associated with handling irradiated materials in reprocessing operations, volatility processing involves more than the usual problems in reagent handling and personnel protection. The highly oxidizing reagents used are hazardous with respect to explosions in the dissolution reaction and also react violently with water (and human tissue).

In the BrF_3 process, all vessels are protected against leakage accidents by secondary containers. Rupture disks are installed in all vessels that might be subject to unusual pressure fluctuations. Materials escaping through the disks would be collected in secondary

process vessels. Accidentally released volatile materials are taken overhead to a scrubbing system consisting of a dilute caustic solution that hydrolyzes the volatiles to aqueous, dissolved fluorides. Volatile process waste and general cell atmosphere are also vented through such a scrubbing system before discharge, and process lines passing through operating areas are of double-wall construction. Exceedingly sensitive leak-detection devices are incorporated in all important process vessels. Operating areas are enclosed to the greatest extent possible and access is permitted only under carefully controlled conditions. Much longer periods of continued safe operation will be required, however, before the safety of the approach can be considered proved.

Low-Decontamination Systems

Two approaches to low decontamination systems are discussed here, aqueous and pyrometallurgical systems. Low-DF systems can only be considered (1) where fabrication is not required, e.g., in liquid-metal or aqueous-homogeneous reactors, or (2) where they are to be accompanied by remote fuel refabrication.

In all low-DF systems for heterogeneous reactors, the problems of processing still include those of developing adequate waste-disposal systems. While the fission products may be in a concentrated, solid form, the problems of off-gas treatment remain, and new ones arise with respect to contaminated cladding, processing crucible and equipment disposal.

Remote Fabrication

The remote-refabrication requirement has many ramifications. One of the more apparent is that the fuel element must be very easily fabricable. In fact, the development of the sodium or NaK bond, because of its relative simplicity, has been closely associated with the increasing interest in remote fabrication. It seems unlikely, at this time, that such elaborate fuel elements as those of the MTR could be economically fabricated by remote means.

In a nuclear-power economy requiring higher fuel burnup and fuel recycling, semiremote fabrication may be required as a minimum measure for any fuel cycle. This involves at least moderate shielding and personnel control. One of the determining factors is uranium-237 buildup caused by neutron irradiation of U^{235}, which is already a limiting factor on the processing of short-cooled uranium materials by conventional aqueous processes.

A comparable problem, though of a somewhat more serious nature, exists in the thorium-U^{233} cycle. The irradiation products of thorium, especially the formation and decay of the thorium-232 chain, may well lead to some form of semiremote handling and refabrication if thorium is to be recycled.

Effect on inventory. Fast reactors on a uranium or plutonium cycle require large core inventories. While 2–4 month cooling can partly alleviate the need for semiremote handling, such a saving is obtained at the cost of even higher inventory charges.

Reactors on thorium cycles may require semiremote fuel handling under any conditions but would certainly require at least as much pre- or post-separation cooling as in the uranium case. For this reason it may be desirable to go to completely remote fabrication, which obviates much or all of the cooling and the associated charges.

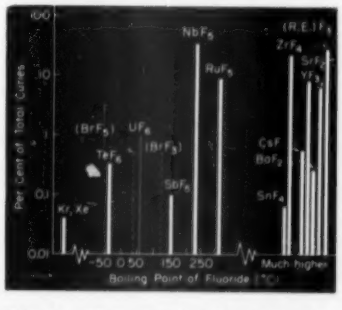

FIG. 3. Fission-product-fluoride volatilities on which volatility processing concept is based. Most are quite nonvolatile relative to UF_6. Serious problem is separation of UF_6 from fluorinating agent (BrF_3). Values are based on 65-day-irradiated, 100-day-cooled U^{235}. (Diagram courtesy of Argonne National Laboratory)

High Processing Cost

The greatest single problem facing designers of low-DF systems is the inherently high unit cost of fuel recovery. Under optimum conditions, one would expect that no transportation of fuel to a processing facility would be required, at least not to the extent involving shipment in shielded casks, etc. Such savings might well become important, but no one has yet been willing to propose the ultimate in such savings—a processing and fabrication facility as an integral part of the reactor building itself, using common storage canals, ventilation, etc. There would seem to be little question but that such arrangements will eventually be used, but at this time transportation to an adjacent site is usually involved using either conventional shipping casks or tunnels (*3*).

However, as of this date, the costs of the processing and refabrication facilities combined are large relative to the average daily throughput of irradiated fuel from a reactor. Presumably, more than one reactor feeding a common processing plant would help alleviate this problem (*4*). Some compromise will eventually be found between the need for high throughput and the transportation and storage factors involved. This problem is of course no different from that faced in the construction of conventional aqueous nuclear-fuel processing plants at this time.

Aqueous Low-DF Systems

At least two low-decontamination-factor systems are now being considered that would use aqueous processing techniques. These are associated with the Homogeneous Reactor Experiment and the as-yet-embryonic Fast Oxide Breeder (FOB).

In discussing these systems, it should be remembered that, in the case of two-region reactors, core processing is aimed at reactivity control while blanket processing is primarily associated with recovery of bred fissionable material.

Fast Oxide Breeder. The FOB (*5*) is a two-region reactor with a core of mixed uranium-plutonium oxides canned in 0.2-in.-i.d. stainless-steel pins. In view of the value of the contained plutonium and the relatively low mass of fuel that must be processed on a daily basis, incentives for consid-

eration of a small, integrated, low-DF plant are strong.

Preliminary research and development on such a concept has already been completed (*5*). An aqueous technique has been proposed in which current solvent-extraction technique would be pared to a minimum. Cold experience with preliminary engineering equipment indicates that it is possible to chop fuel pins, completely dissolve and leach the contained mixed oxides and transfer to modular, disposable, solvent-extraction equipment in which partial decontamination can be obtained. The chop-leach technique, because of its early separation of diluting inerts, permits the attainment of maximum throughputs in minimum-sized equipment. The product material can then be precipitated, fired, extruded and sintered as a preliminary to loading the fuel tubes.

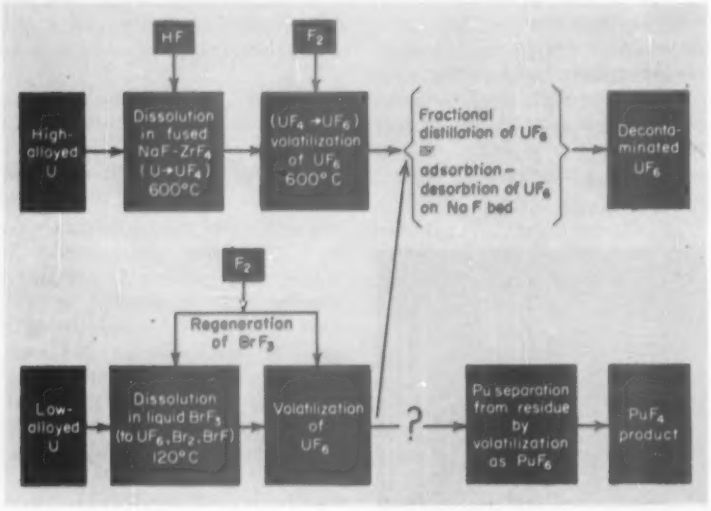

FIG. 4. Alternate volatility processes. High-alloyed fuel (highly enriched U) is fluorinated in two steps at high temperature. Low-alloyed fuel is processed at lower temperature to maintain liquid BrF_3

The FOB blanket, probably consisting of depleted uranium, would be processed in conventional equipment after transportation to a central plant. This choice is based on the relatively low value of contained material and the larger quantities to be handled.

Homogeneous reactor. Processing for the Homogeneous Reactor Experiment (HRE-II) is a composite of standard and new techniques (*6*).

In brief, the homogeneous, heavy-water–uranyl-sulfate core is continually processed. Significant thermal poisons, such as rare earths and corrosion products, are removed by capitalizing on the limited solubility, and in fact the retrograde solubility in some cases, of their metal sulfates and oxides.

With small hydroclone separators, it is possible to separate the dilute slurry formed in the reactor into two streams, one quite concentrated and one very dilute, the latter being returned to the reactor core. The heavy water is then recovered from the concentrated stream, and the solids, including part of the uranium removed with the slurry, are sent to a conventional solvent-extraction plant. The quantity of material processed per megawatt-day is, of course, relatively small because of the concentrated nature of the fuel (U^{233}) and the relatively small amount lost per processing pass.

The blanket, probably a slurry of thorium oxide in heavy water, is processed using modified Thorex techniques. In practice, it would probably be combined with the solid discharge stream from the core.

Pyrometallurgy

The pyrometallurgical approach to low-DF processing includes a number of methods (*7*). In addition to separation based on oxidation and reduction (table, p. 46), separations may be based on fractional distillation, e.g., plutonium separation from uranium by volatilization of plutonium at 1,600° C; separation by crystallization, such as in the dissolution of uranium in molten zinc or mercury followed by a selective precipitation or separation of uranium-solvent intermetallics; and liquid-liquid extraction methods, such as the separation and concentration of bred plutonium in irradiated uranium by contact with molten magnesium.

All such processes have certain inherent advantages in addition to those associated with low-decontamination processes in general. These are:

1. Irradiation effects on process reagents are considerably less severe than in solvent extraction.

2. Waste-disposal operations are simplified due to the more concentrated form in which the fission products are discharged from process streams.

3. Criticality problems are lessened by absence of large quantities of moderating materials.

4. The processes are inherently compact. (In an extreme case, a volume reduction of 80:1 is observed in comparison with a process using uranium aqueous solutions.)

Because of the wide range of possible techniques required if a process is to be "tailored" to a specific reactor concept, it will be appropriate to examine two cases—the Experimental Breeder Reactor (EBR-II) and the Liquid-Metal-Fuel Reactor (LMFR).

EBR-II. The EBR-II, a fast reactor, is perhaps a perfect case in point. The core of the EBR-II will initially consist of highly enriched uranium alloy material clad in stainless steel with NaK as a bonding, heat-transfer medium. An element will consist of a large quantity of small fuel pins.

A fast reactor is peculiarly suited to pyrometallurgical processing and was the first type for which such a process was proposed. This suitability arises from the relative insensitivity of fast reactors to moderate concentrations of fission products that would be significant poisons in thermal reactors.

In the EBR-II, as with fast reactors generally, the core inventory of fissionable material is very high and as such there is a premium on minimizing the amount of material tied up in the processing-refabrication cycle. Actually, the difficulty of separating completely the low electropositive elements from the reprocessed uranium could be an advantage. Recent experimental evidence indicates that these elements assist in maintaining the structural in-

tegrity of high-U-alloy fuel elements under irradiation.

The proposed processing scheme for the core follows rather closely the oxidation-reduction approach mentioned earlier. Magnesia or zirconia crucibles will be used for processing, the wall material of the crucibles themselves serving as the oxidizing agent.

To control build up of the low-electropositivity fission products, a bleed-off of core material will be required. If they build up too fast, a recently devised technique (*7*) for their removal may be considered. In it, molten zinc is used as a dissolvent for the uranium-fission product stream. By temperature adjustment, it is possible to separate a relatively pure U-Zr compound from which zinc can then be removed by distillation.

With continued operation of the reactor, the initial charge of U^{235} will gradually be replaced with the Pu produced in the breeding blanket. However this will not seriously affect the core reprocessing operation due to the relative positions and similarity of uranium and plutonium compound stabilities as shown in Fig. 5.

Present thinking is that plutonium bred in the blanket can be separated by a liquid-liquid extraction technique. At present, this is envisioned as an extraction of plutonium from a molten 5 w/o chromium-uranium blanket alloy. Molten magnesium is a satisfactory extractant and can be subsequently distilled away from the plutonium.

LMFR. The LMFR uses a core of uranium dissolved in molten bismuth. Inasmuch as it requires no refabrication, it represents the ultimate and most obvious application of pyrometallurgical reprocessing.

In this reactor, which will use the thermal thorium-uranium breeding cycle, neutron economy is again quite critical because of the relatively low number of neutrons produced per fission. The continuous removal of the fission products, especially xenon and samarium, can significantly decrease the over-all thermal poisoning. The xenon also adversely affects the over-all negative temperature coefficient of reactivity, which is desirable from a reactor-safety viewpoint.

Two processing possibilities are currently under investigation, both of them unique in that they envision continuous processing of the reactor core. Both systems propose xenon removal by inert-gas sparging. The fission products more active than uranium can be extracted by contacting the uranium-bismuth solution with an exactly equivalent amount of uranium or bismuth chloride, these oxidants being carried in a mixture of inert salts. All fission products more reactive than uranium are transferred to the salt phase; the uranium or bismuth that has been displaced from the salt enters the uranium-bismuth core solution, thus either diluting or enriching as may be desired. Such a system requires careful concentration control, and in practice provision for recovery of uranium chloride from the salt phase would be required. A variation has been proposed in which such fine control is not essential.

An entirely different scheme has recently been suggested for the core of the LMFR. It is a combination of conventional oxidation-reduction technique with electrochemical and volatility processing. In this case, the core material would be contacted with a fused-salt mixture that has previously been saturated with hydrogen fluoride. Under controlled temperature conditions, the uranium and the more active fission products would be converted to fluorides and dissolved in the fluoride salt medium. This treatment would be on a batch basis.

The molten salt mixture would be sparged with fluorine as before and the volatile uranium hexafluoride treated with an excess of hydrogen gas and then introduced as uranium tetrafluoride into a clean fused-salt bath. The previously stripped bismuth would now be used as a flowing metal cathode into which the uranium could be electrolytically reduced after which the bismuth stream would be returned to the core. It can be seen that this processing proposal utilizes several of the nonaqueous approaches.

Thorium and ceramic fuels. In the event that thorium or perhaps ceramic fuels become more popular in future heterogeneous reactors, the applicability of pyrometallurgical reprocessing is somewhat less clear.

As has already been pointed out, the extensive use of thorium fuels implies the development of at least semiremote fabrication techniques. The obvious extension of pyrometallurgical methods to such fuels has long been considered but has been regarded until recently as somewhat unlikely because of the much higher melting point of thorium metal and thus the presumed higher operating temperature required for processing. Recent developments, however, indicate excellent potential or electrochemical separation methods operating at lower temperatures, perhaps in the area of 700° C.

Another possibility that has been suggested is the utilization of one of the lower melting thorium-metal binary alloys, e.g., the thorium-magnesium system (*8*). By proper adjustment of temperature and concentration variables in this system, it is thought possible to separate essentially pure uranium from the thorium metal in which it has been bred. After separating the magnesium, the thorium would presumably still have to be cast at its melting point.

The application to ceramic fuels is somewhat less obvious. In the oxide system very preliminary work has indicated a possible technique, however. Under certain conditions, uranium oxide can be reduced with a magnesium-zinc mixture (*9*). Conceivably, the uranium metal could now be separated in the usual way and then reoxidized to UO_2. At this early stage, however, no thought has been given to problems of refabrication of oxide elements other than those for the FOB.

* * *

In addition to the references cited, the author has drawn heavily on other papers presented at the Symposium on the Reprocessing of Irradiated Fuels held at Brussels, Belgium, May 20–25, 1957. The symposium proceedings were published as TID-7534 by the U. S. Atomic Energy Commission.

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Remote Fuel Fabrication —How It Was Done for EBR-II

Remote refabrication of irradiated fuel might lower fuel-recycling costs by permitting shorter cooling periods and simple reprocessing. Such a fabrication system has been designed and engineered for EBR-II

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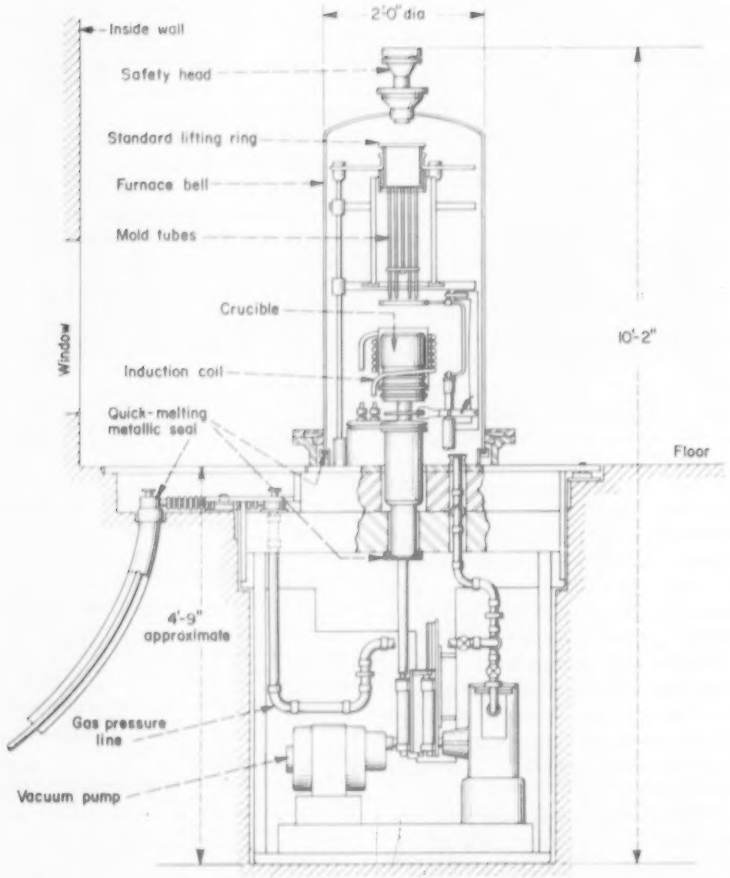


FIG. 1. Injection-casting furnace for making EBR-II fuel pins. With furnace under vacuum, crucible containing melted fuel material is raised on ram until metal covers bottom of mold tubes. Ram actuates switch, which turns off vacuum system and turns on pressure system to force metal into molds

STUDIES OF REACTOR ECONOMICS indicate two cost-reduction possibilities that have not been fully exploited—the reduction of fuel-inventory costs by shortening or eliminating the cooling period required prior to processing and the reduction of high costs of decontamination to human-tolerance levels. To achieve both goals, it has been proposed that the fuel be processed only enough to remove major fission-product poisons.

We consider here some practical aspects of the remote-fabrication methods for such partially decontaminated material. The fabricating system developed for the EBR-II fuel rods, which is based on these considerations, is also described.

Basic Considerations

The development of remote-controlled methods of reprocessing and refabricating reactor fuels must be justified on an economic basis. Therefore, any such process will have the following aims:

1. Reduction of the total fuel inventory by the elimination of, or shortening of, the cooling cycle prior to processing.

2. The elimination of the chemical-recovery plants required to bulk process the reactor fuel.

3. The development of simplified processes that will reduce the cost of both equipment and manpower.

4. Reduction of handling costs and the elimination of shipping costs by

processing and refabricating the reactor fuel at the reactor site, perhaps within the primary reactor shield.

5. The reduction of radioactive-waste storage costs through the production of small-volume, dry wastes rather than large volumes of fluid wastes.

With a process removing only part of the fission elements, the balance of those elements not removed will achieve a stable equilibrium composition after several cycles of equal irradiation and reprocessing. It has been our experience that the fission-alloy compositions are hard and somewhat brittle and are generally difficult to work by conventional methods. The fabrication process must be capable of producing precise shapes from this "accidental" alloy that will meet the metallurgical and physical requirements of the reactor.

As higher burnup of the fuel alloy is achieved with shorter reactor cycles, the radiation levels in any process cell will become progressively higher. The problems of the fission alloy, of fabrication under conditions of intense irradiation by semiautomatic equipment, and of remotely operating, maintaining and possibly replacing this equipment require a fresh approach to the practices of metal processing and equipment design. Both the processes and the equipment must be greatly simplified.

Incidentally, one of the more difficult features of equipment design for remote operation is a human one—to get designers to forget those design conventions that are entirely imposed upon machinery by the limitations of human anatomy. It is for this reason that manipulators and equipment continue to be designed that look and function like similar equipment designed for manual operation.

The "Hot" Material

Each gram of a typical fast-reactor fuel, cycled at ~1½% burnup and cooled for 15 days before reprocessing, will radiate gamma rays approximately 40 times as intense as one curie of radon. Pyrometallurgical processing (see p. 46 of this issue) may reduce this level to ⅕ or ¼. If reasonable-sized batches are to be processed, these are extremely intense radiation sources.

The radiation one foot from a 10-kg batch of this fuel will have an intensity of 2×10^5 r/hr. The background radiation within the work areas of a process cell handling this material will be 10^4–10^5 r/hr with local zones of much more intense radiation.

Massive pieces of this material or compact assemblies of reactor fuel elements are intensely heated by self-absorbed radiation. The total heat generated increases not only with the increase in mass but also with increased absorption as larger masses are assembled. A 6-kg assembly of small rods in close-packed array with a packing density of approximately 50% generates heat at a rate of ~0.4 watts/gm. Forced cooling must, therefore, be provided whenever massive quantities of the metal are worked on. The heating effect also makes an inert atmosphere necessary to protect surfaces and to prevent metal fires.

Lubricating oils are damaged by 10^8–10^9 roentgens. Provision must be made to pump the oil through the bearings or, where grease is used, the bearings should be so designed that incoming grease flushes out the old grease. Unfortunately, certain of the components of oils and greases tend to cross-link and to harden while others are cracked to volatile substances. This creates a particular problem in the lubrication and sealing of mechanical vacuum pumps. Equipment to operate in the immediate vicinity of the process material should work without lubrication or with a dry lubricant.

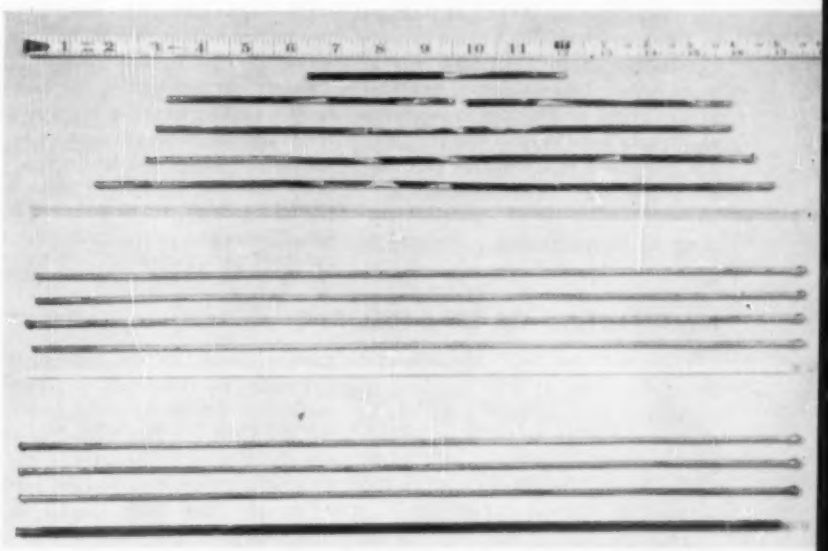


FIG. 2. Uranium castings made by injection-casting. Castings were made in graphite-washed glass molds (bottom group), magnesium zirconate-washed molds (center) and unwashed molds (top). Uranium was cast at 1,340° C into molds at 450° C and 45 psi. Best results are obtained with graphite-coated molds

Remote-Equipment Design

The high radiation levels make radiation damage a problem to be considered in the design of process equipment used in the process cell.

Organic materials. Rubber, neoprene and polyethylene are damaged at 10^7–10^8 roentgens and, in our process cell, with 10^5-r/hr background, will last only a few hours. Plastic seals on vacuum equipment, bearings, gasketing for pipes and vessels must, therefore, be avoided or provision must be made to replace them at intervals.

Optical glass. Even the best non-browning optical glass will lose transparency in short periods of time. Optical systems exposed directly to the fuel, such as inspection periscopes and microscopes, should be mirror optics. It may be necessary to equip windows with shield shutters to prolong the useful life or to make the front layers of the windows replaceable.

Electrical equipment. Conventional electrical insulation will be damaged in a few hours in the high radiation environment. Mineral-insulated conductors and cables of the type using a compacted magnesia insulation in a swaged copper or steel jacket are satisfactory, provided a way may be found

to pot the ends without the use of organic materials.

Conventional electrical motors are not satisfactory unless they can be shielded. It is possible that the new class C motors will be satisfactory from the radiation standpoint once operational bugs have been worked out. Some experimental work has been done in the design and construction of electric motors and transformers using aluminum conductors with a heavy anodized coating insulation.

The highly ionized atmosphere in the process cell will no longer be a good insulator. Difficulties with switches, commutators, and slip-rings from sparking and corona effects can be avoided by using multiphase-induction motors. Corona-like effects have been seen at much lower voltages than will be visible in air. Very preliminary experiments at 10^6 r/hr have indicated that it may not be possible to operate the electrical equipment at 440 volts although both 220-volt and 110-volt operation appears to be satisfactory. Because of its higher ionization potentials, helium may be preferable to argon as the cell atmosphere.

Atmosphere purification. Continuous purification of the atmosphere is necessary to compensate for the air introduced with materials and for minute leakage into the system.

Oxygen can be removed from argon by catalytic reaction with hydrogen and adsorption on driers. The argon can be purified of the other gases by liquifaction and fractionation. Helium can be purified by adsorption of oxygen, nitrogen and other gases upon activated carbon or upon another adsorber such as silica gel, alumina or the synthetic zeolites. The total impurities in the helium can be reduced to below 100 ppm by this method.

Safety requires that the process cell be at a slightly negative pressure with respect to the room atmosphere to ensure that leakage will be inward. In order that the purifying system need not be too large, the process cell must be as leak tight as possible.

MANUFACTURE OF EBR-II FUEL PINS

The fuel for the Experimental Breeder Reactor II is in the form of cylindrical rods 0.144 ± 0.001 in. in diameter and 14.22 in. long. These are enclosed in stainless-steel tubes with a 0.006-in., sodium-filled annulus between the fuel rod and the tube.

After irradiation, the pins are removed from the tubes and pyrometallurgically processed to remove the volatile and more easily oxidized or slagged constituents. The composition of the processed equilibrium alloy is given in the table.

A uranium-base alloy of this composition was found to be hard in the as-cast condition (46–48 Rockwell C). It cannot be worked in the range of 300–600° C as can uranium. It can be reduced by rolling at 800° C. Above this temperature, the metal appears to be somewhat brittle.

Several methods were considered for fabricating this fuel. Conventional wire fabrication was ruled out because of the lack of ductility and the complexity of the process.

Powder metallurgy was considered. However, we looked with distaste at the prospect of handling many kilograms of thermally hot, explosively pyrophoric, intensively radioactive metal powder in the process cells.

Finally we developed a successful method of precision-casting fuel slugs for the EBR-I. The EBR-I castings are about $\frac{3}{8}$ in. in diameter by $3\frac{3}{4}$ in. long with a diameter-to-length ratio of 1 to 10. The long slender shape (diameter-to-length ratio of 1 to 100) gave practical difficulties in the centrifugal casting process and the machine required was somewhat complex for remote operations.

Process Requirements

A fresh approach was needed. Several requirements were apparent:

1. The process should have a minimum number of operations.
2. Subassemblies must be replaceable as units with a minimum of assembly and adjustment in the cell.
3. Routine operations should be mechanized. Manipulators should be used to service the equipment and to perform nonroutine operations.
4. Mechanisms, locking devices, fasteners and clamps should be designed for simplified one-movement operation adapted to the manipulator capabilities.
5. The construction materials should be radiation resistant.
6. Auxiliary equipment should be shielded from direct exposure to th radioactive fuels so that vacuum pumps, valves, mechanical drive units and electric motors may operate under low-level radioactivity conditions.
7. The equipment should be designed for total replacement to allow complete flexibility of processes.
8. Since the only really powerful movers available within the process cell are bridge cranes and vertical-lift mechanisms, all equipment should disassemble in the vertical direction.

Precision Casting Selected

Precision casting was selected as the method of greatest promise. The method proposed was a simplified nonmechanical injection-casting process using the pressure of an inert gas (helium or argon) plus the inertial effect of the moving metal to fill the mold. This method was as follows:

1. A shallow crucible of metal is melted under vacuum in a vacuum-pressure furnace. A cluster of tubular molds, with the top ends closed, is suspended above the crucible with the open ends pointed down toward the metal bath.
2. The crucible is raised, immersing the open tube ends in the liquid-metal bath and the furnace is simultaneously pressurized with inert gas to several atmospheres.
3. The gas pressure forces the molten metal into the mold cavities.

A schematic drawing of the equipment is shown in Fig. 1. The furnace bell is sealed to the furnace bottom by a solder-freeze seal. It is held against the internal pressure by a cam-acting clamping ring and is designed to withstand an internal pressure of 11 atmospheres. It is mounted on a 1-ft-thick shield block, which rests in a recess in the process-cell floor above a shielded pit area. The furnace bell is convection cooled so that its temperature does not normally go above 200° C.

The crucible is mounted on a vertically acting ram, which in its lower position centers the crucible in a solid-conductor induction coil and which may be raised to either of two positions above the induction coil by the action of a double-slide pneumatic cylinder assembly located below the shield block. The ram acts as a solid shielding plug where it passes through the shield block. It is vacuum sealed below the shield where it is protected from direct radiation exposure.

The shielded-pit area houses all equipment of questionable radiation resistance, including the vacuum pumps, electrical components, solenoid-operated valves, accumulators, lubricated gear reducers and other equipment requiring organic or radiation-sensitive materials. This equipment is mounted on a frame, which is attached to the bottom of the shield block and is lifted out of the pit with the shield block for servicing or maintenance. The equipment pallet disengages from the supporting frame and each item of equipment may be lifted free by the operation of simple cam keys.

The molds are nonpermeable refractory tubes closed at one end and suspended open end down above the crucible in a $4\frac{3}{4}$-in.-diameter holder that can hold a maximum of 99 rods. The precision-bore, high-silica-glass tubes are coated on the inside with a colloidal-graphite wash. These are commercially produced as precision-bore tubing with a bore tolerance of ± 0.0005 in. Vycor tubing was found to have low resistance to mechanical shock. The wash, colloidal-graphite or thoria, separates the metal from the silica. Where the wash coating is imperfect, a reaction layer forms between the silica and the metal, which causes a slight roughening of the casting surface. Where the coating has been applied uniformly, the cast surface is extremely smooth and glass-like. Tolerances of ± 0.001 in. diameter are produced in approximately 96% of the castings. Castings are shown in Fig. 2.

The zone of the furnace holding the mold tubes is independently heated by a resistance heater. The molds are shielded from direct heat radiation from the crucible by a tantalum shield, which automatically swings out of the way as the crucible is raised. Power for the induction coil is supplied by a conventional high-frequency generator outside of the shielded area.

The charge is melted under high vacuum as supplied by a modification of a conventional high-vacuum pumping system. Since the molds are within the vacuum chamber, they are also evacuated. The high-vacuum system is valved off immediately before pouring, and the furnace vacuum is maintained by the mechanical pump alone. Pressure is supplied from an accumulator tank.

A two-way selector valve connects the furnace either to the mechanical pump or to the accumulator tank. The excursion of the ram upward raises the molten-metal level above the lower end of the tubes and trips a snap switch, which moves the selector valve from the vacuum system to the pressure system. The sudden increase in pressure forces the alloy up into the tubes.

Calculated Equilibrium Composition*†

Element	%
Zirconium	0.10
Niobium	0.01
Molybdenum	3.42
Technicium	0.99
Ruthenium	2.63
Rhodium	0.47
Palladium	0.30
Total	7.92

* 5 w/o new metal. Elements present in quantities >0.01% are listed.

† Composition values, calculated by I. G. Dillon, are taken from an Argonne National Laboratory Internal Memorandum dated Sept. 11, 1956.

An analysis of the gamma-radiation heating effect has indicated that the number of pins to be cast at one time will be limited. The central-pin equilibrium temperature of a furnace pallet of 99 pins was over 1,400° F. The furnace operation will be a batch operation and, contrary to the third preliminary requirement, charge preparation, loading and unloading of the furnace will be by manipulator.

After cooling, the castings, while still in their tubes, will be transferred to an inclined feeder. This device will feed the pins between two concave, roughened rolls, which break the glass into small fragments away from the castings.

The castings will then move out beneath an inspection periscope. A selector device operated from outside of the cave will move the castings either to the scrap chopper or on to the cutting operation.

The acceptable castings will pass into a double-blade shear, which will simultaneously crop top and bottom of the casting, leaving a pin 14.22 in. long.

After shearing, the castings will be moved by the transfer device through the inspection station, where they will pass through pneumatic gaging devices measuring their diameter and length, across the suspension of a semi-automatic balance, where they will be weighed to the nearest 10 mg and finally under a reflection optics periscope, where the surfaces will be inspected for porosity, laps or other conditions that would interfere with liquid-metal bonding. The fuel-element castings will then be inserted into a rotary-magazine of fuel tubes, which will fit the tube-welding machinery for the jacketing and bonding operations.

Conclusions

While the foregoing sequence of operations is much simpler than is required for conventional wire-making, we feel that it should be still further simplified. Experiments are under way on casting the fuel directly into refractory-metal jacket tubes and forming a direct diffusion bond layer between the tubes and fuel alloys.

Other reactor cycles have been proposed in which the fuel is refabricated by more conventional equipment within hot process cells. It has been proposed that plate-type fuel assemblies be fabricated by conventional ingot casting and rolling practices using conventional rolling mills with special adaptions for remote operation. I believe that this can and will be successful but that a great deal of attention should be given to the design of the equipment so that it can be maintained in operation. A pre-analysis should be made of the operations of removing and reinstalling every part that may require maintenance or replacement. It would be highly desirable for the equipment designers to actually carry out some of the operations in repairing equipment by remote-control procedures in order that they may experience some of the limitations of working under these conditions. Often, very simple manual operations are impossible to do by manipulator, requiring a basic change in the equipment design.

The foregoing remarks are not intended to discourage the designers of power reactors from considering remote reprocessing and refabrication as a method of lessening the cost of the reactor fuel cycle. We believe that it is a necessary part of the fast-reactor fuel cycle. As satisfactory methods of reprocessing and refabricating are developed, it may become a part of the thermal-reactor cycle.

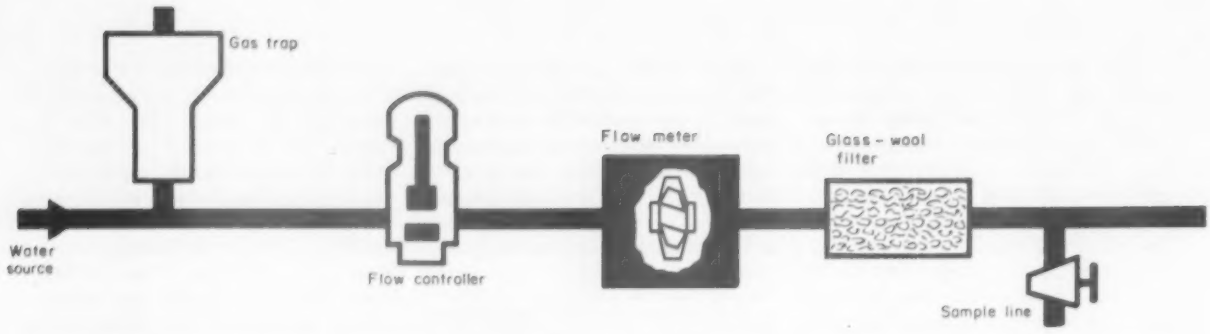


FIG 1. Iodine-separation monitor has cation resin to remove solids and long-lived activities. Discrimination against low-level

Fission-Product Monitoring

At the MTR burst fuel is detected by recognizing fission-product iodine. Filtering through a cation resin and low-energy-gamma discrimination makes the test specific for iodine isotopes

By R. L. HEATH
Phillips Petroleum Co., Idaho Falls, Idaho

IN HIGH-FLUX, WATER-COOLED reactors detection of fission-product activity in coolant streams is an important method of finding fuel-element failures. MTR experience has indicated the limitations of systems normally employed for this purpose. This has led to the development of a new fission-product monitor exhibiting nearly ideal characteristics. It uses continuous isolation of fission-product iodine by ion-exchange resins and gamma-energy discrimination (Fig. 1) and thus responds essentially only to 86-sec I^{138}, 54-min I^{134}, and 6.7-hr I^{135}.

This offers the advantage of unique response to short-lived fission products. Thus the system measures the instantaneous production rate, and response time is limited only by transit in the system. The use of iodine activity integrated on an anion resin column achieves sensitivity orders of magnitude above direct-stream observation. Operational experience with this type of monitor has indicated negligible response to non-fission-product background activity and more than adequate sensitivity for the detection of fission products.

Other fission-product monitors that we have developed are discussed in the box on p. 56. They include a delayed-neutron monitor and a differential gamma monitor using direct-stream gamma detection with energy discrimination. The differential gamma monitor is less sensitive than the iodine monitor, but it does not require chemical-separation techniques. The delayed-neutron monitor is not sensitive enough for routine detection of fission products, but because it detects short-lived emitters it may be particularly suitable in some reactors for locating the source of fission products once their presence is detected.

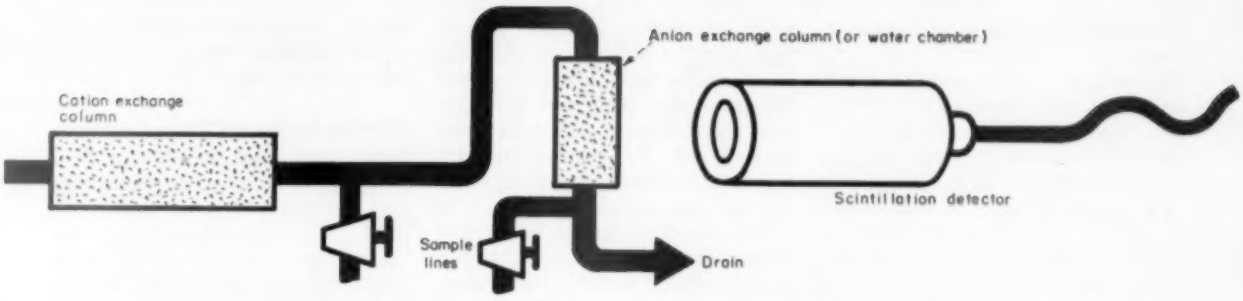
Monitoring Problem

The two major problems encountered in monitoring for fission products in a reactor coolant stream are (a) the detection of fission products and the determination of the instantaneous production rate in the system; (b) the location of the source of these fission products. The best solution to one of these problems is not necessarily applicable to the other. In a reactor such as the MTR with high-integrity fuel and a core composed of a small number of elements, the second problem is not so formidable. For this reason our emphasis has been on the answer to the first question—are fission products present in the coolant?

Background radiation. The problem is best described in terms of a "signal-to-noise ratio," the noise being the radioactivity due to both irradiation of trace amounts of material introduced into the water stream and the entrance of radioactive nuclei into the stream from structural materials. This problem is further complicated by deposition of radioactive material on the surfaces of the water systems, which can result in contamination of stream-type monitors. The sensitivity of any detection system that relies on an indication of radioactivity associated with fission products will depend then on the ratio of gross-fission-product activity to background unless some means is provided to increase the relative sensitivity to fission products.

An examination of possible criteria for positive identification of fission-product activity in the presence of background indicates the following possibilities:

- Detection of delayed neutrons
- Gamma-ray spectrometry
- Detection of high-energy fission-product beta particles or Cerenkov radiation



gammas limits effluent count to activity of fission-product iodine. Anion column accumulates iodine. Result is sensitive system for . . .

in Reactor Coolant Water

• Chemical isolation of one or more fission products, on either a batch or continuous basis

• Detection of particulate matter where fission products can be assumed to be solid

• Detection of gaseous fission products

Our methods make use of three of these effects.

Fission-Product Activities

To understand the response of any monitor to the addition of fission products, the general character of fission-product radiation must be considered. The relative abundances of different nuclides at the point of detection depend on the previous irradiation history of the fissionable source material, the time delay after irradiation, and the fission yields for the various nuclides under consideration. For our purpose it was sufficient to determine the general character of gross-fission-product radiation as related to the performance of the various possible monitoring systems.

To establish the feasibility of the possible methods for improving the signal-to-noise ratio in fission-product detection, the following major points have been investigated:

• Methods for radiochemical isolation of fission products.

• Gamma spectra of short-lived fission products to be expected with the time delays required for operational monitoring systems.

• Fission-product evolution from fuel after exposure to the water stream.

• Distribution of fission-product activities in the water system after entrance into the stream.

After considering several possibilities among the major fission products, we feel that the most favorable monitoring method is detection of the iodine fraction since:

◆The chemical procedure for separation of iodine from gross fission products is simple and reliable with high

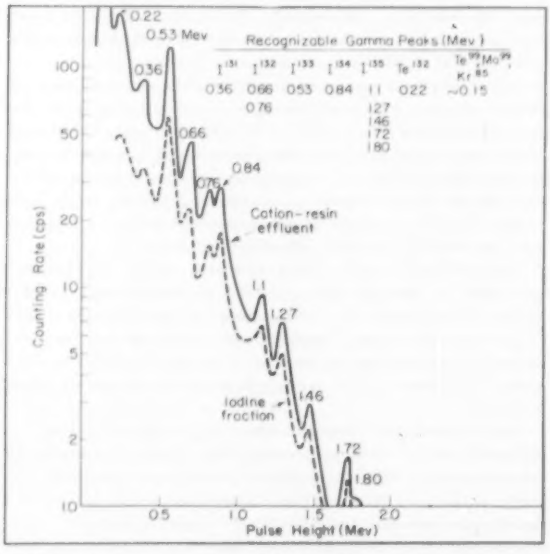


FIG. 2. Gamma spectrum of iodine fraction separated from irradiated uranium (dotted) is closely similar to that of cation-resin effluent (solid)

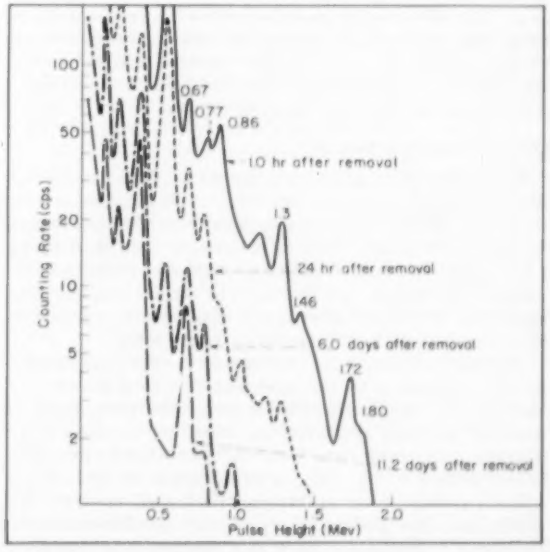


FIG. 3. Spectrum from anion resin. Its decay is used to establish irradiation history of fission-product source after a fuel rupture is discovered

decontamination factor and chemical yield.*

◆Identification of the presence of any of several iodine isotopes is a unique indication of fission-product activity.

◆Iodine activities of interest are high on the fission-yield curve.

◆A spread in half-lives from 86 sec to 8 days offers some possibility of analysis to determine previous irradiation history.

◆Physical properties of iodine insure its release under all proposed mechanics associated with fuel-plate rupture.

◆Iodine is not absorbed or plated onto the metal surfaces of the reactor water system.

Iodine Monitor

To gain a thorough understanding of the fission-product iodine activities, we made gamma spectroscopy studies of the gross fraction as well as the individual isotopes and learned about the general distribution of radioiodine and its precursors in the reactor water system.

Figure 1 is a block diagram of the monitoring system presently used at the MTR. A side stream from the main reactor coolant loop is passed through a glass-wool filter† to remove solid material and then through a cation resin bed. In the aluminum MTR system the cation bed removes virtually all activity in the absence of fission products. In the presence of gross-fission-product activity, the cation-bed effluent contains only fission-product molybdenum, technetium, tellurium, bromine, iodine, and gas activities. This offers a unique indication of fission-product activity in the coolant stream.

Several alternative methods are available for detecting fission-product gamma activity in the cation effluent. These include the detection of gross activity in the water stream or in the materials absorbed in an anion column used to increase sensitivity. In the accumulated-activity monitor shown in Fig. 1, the cation exchanger is followed by an anion resin bed. The anion bed can be replaced by a water chamber if accumulation is not desired. The anion bed or chamber is then viewed by a gamma detector with a suitable shielding arrangement to reduce the background.

Activity Accumulation

The advantage of observation of accumulated activity on an ion column is that it affords detection sensitivities for fission products orders of magnitude above those to be expected by direct-stream measurement. This method is worthy of consideration in cases where dilution factors are large yet high sensitivity and selectivity are major considerations.

Response time. The half-lives of the detected nuclides and the transit time in the stream are the most important factors in determining response time. Observing the nuclides collected by an anion column under the conditions described, we see a distribution of half-lives extending downward from I^{131}'s 8.0 days. The lower limit for response time is established by the transit time to the column. Inspection of the gamma spectrum yet fission products collected by the anion column and the spectra of the separate activities shows that energy discrimination will allow rejection of the long-lived activities.

* L. E. Glendenin, R. P. Metcalf in "Radiochemical Studies: The Fission Products," C. D. Coryell, N. Sugarman, eds., (McGraw-Hill Book Co., New York, 1951).

† Note added in proof: The glass-wool filter has been replaced by a paper filter in a later version of the apparatus.

Two Other Methods for Fission-Product Detection

In addition to the iodine-separation method, two others have been developed for fission-product monitoring. The differential gamma method is usable in test loops where chemical-separation techniques are inconvenient. The delayed-neutron method will be useful when locating the source of fission products becomes as big a problem as detecting their existence.

Direct Gamma Monitor

The differential gamma monitor of Fig. 4 is free of chemical-separation complications. It consists of two detection channels and an output that is proportional to their difference. One channel responds essentially to N^{16}, which dominates the background count in the absence of fission products. The other channel responds to that portion of the fission-product spectrum that offers the greatest signal-to-noise ratio.

Development studies. **To investigate the possibilities of this approach to the problem it was necessary to investigate the general character of the gross-fission-product gamma spectrum at short times after irradiation. To establish realistic conditions for this investigation it was decided to simulate an alloy-fuel-plate rupture with MTR-type fuel and observe the fission-product gamma spectrum in the effluent water stream. A fission-product test facility was designed for operation in a high MTR flux. Rupture conditions were simulated by inserting a ¼ x ⅛-in. section of unclad MTR fuel plate into the reactor with means provided for external observation of coolant-water flow past the section. A second, identical tube containing no fuel was provided to allow a direct comparison with the normal MTR process-water activities.**

Spectral data were obtained for delay times varying from 10 sec to several minutes by variations of the holdup time of the system. The fuel section employed had been irradiated in the reactor for several hours. The shape of the gross spectrum of the process water for short delay times, predominantly due to N^{16}, is independent of delay for times up to 1 min. Thus one can use the N^{16} level as a background signal.

Comparison shows a maximum ratio of fission-product to background activity in the energy region from 0.7 to 1.5 Mev—nearly independent of delay time. Thus, by observing gamma radiation in this region, some improvement is obtained in the signal-to-noise ratio over that to be expected by observation of gross activity.

Disadvantages. **Among the major operational disadvantages of this approach have been response to abnormal non-fission-product activity and gradual contamination of the water chambers with long-lived background activities requiring frequent service. The foremost objection, however, has been the electronic complexity of the basic approach. The equipment necessary is complicated, and adjustments of the**

If a simple level discriminator is employed to reject all pulses due to gammas below 800 kev, this limits response to the high-energy gammas emitted by 6.7-hr I^{135} and 52-min I^{134} with some contamination from the low-intensity gammas of 2.4-hr I^{132}. There also remains the possibility of detecting the 1.4- and 2.8-Mev gammas from 86-sec I^{136} if the delay is reduced enough.

The decay of gamma radiation above 1 Mev from activity collected on an anion column shows a major component due to 6.7-hr I^{135} and a lesser one attributed to 77-hr Te^{132}-I^{132}. No activities of longer half-life are observed under these conditions of operation. In this case the column had been operated with a time delay estimated in excess of 10 min.

Rdiochemical Studies

An extensive series of experiments preceded the development of the iodine monitor. To provide the background information for the use of fission-product iodine as a radiochemical index, iodine fractions of gross fission products were prepared by CCl_4 separation from samples of uranium irradiated in the MTR. Gamma analyses were used to check purity as well as the relative isotopic abundances of the various iodine activities present.

Figure 2 includes a gamma pulse spectrum obtained on a scintillation spectrometer from a sample of the iodine fraction. Successive spectra are used to show the decay of the gross iodine fraction.

On the basis of half-life and gamma energy it is possible to recognize the presence of I^{131}, I^{132}, I^{133}, I^{134}, I^{135}, and Xe^{135}. The absence of any gamma activities not associated with these iodine isotopes or their daughters indicates the high level of purity obtained with the simple CCl_4 chemical separation.

Relative abundances of different iodine activities as a function of time in any given sample containing gross-fission-product activity will be determined primarily by the fission yields of the mass chains and the nuclear properties of the members of each decay chain.

Irradiation history. If time relationships are known and the gamma spectrum of each isotope can be deter-

Direct Gamma Monitor—Another Method

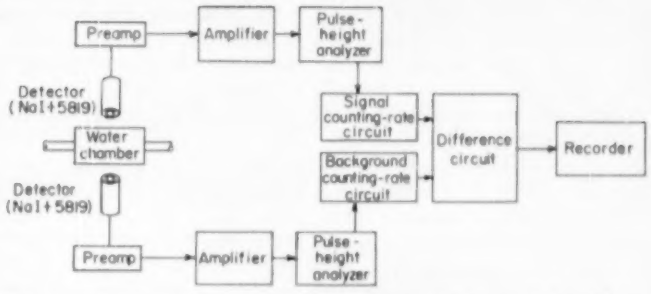


FIG. 4. Direct-stream gamma monitor compares background signal from N^{16} with fission-product count in 0.7–1.5-Mev region (see box, p. 56)

equipment to ensure proper operation are not entirely straightforward.

Delayed-Neutron Monitor

Escaped fission products can also be determined by the detection of delayed-neutron activity if the delay time from production to detection is sufficiently short. The feasibility of this method was studied along with the simulated-rupture studies.

Delayed neutrons accompanying fission have been separated into at least six components varying in half-life from 0.05 to 55.6 sec. Thus we must use quite short transit times to permit detection of a maximum number of delayed neutrons. Any practical monitor requires a compromise on delay and holding times to optimize the signal-to-noise ratio.

The major source of background is photoneutrons from the γ,n reaction on deuterium in the water and other hydrogenous material near the neutron detector. This problem is worsened in heavy-water-cooled reactors. The responsible gamma emitters are those present in the water stream with gamma energy above the deuteron photodisintegration threshold (2.62 Mev): primarily 7-sec N^{16}.

Sensitivity of this method was explored by measuring neutron activity from the water chambers on the rupture-test facility using a $1\frac{1}{2}$ x 10-in. BF_3 counter. The counter was surrounded by a 9-in.-dia cylinder of paraffin moderator covered with Cd sheet on all sides except that toward the water chamber. Although the neutron counting rates were less than the gross-fission-product activity observed with the gamma detector just described, the signal-to-noise ratio was considerably better. It was concluded that this type of system would be quite satisfactory for the detection of fission products under the experimental conditions of this test. However, the sensitivity would be highly dependent on delay time because of short half-lives.

Evaluation. **The two known delayed-neutron emitters, Br^{87} and I^{137}, are anion forms and would be accumulated on an anion exchange column. This would indicate that the detection efficiency could be increased by the addition of an anion exchange column.**

Even so, in a reactor of the MTR type, the delayed-neutron system rates a poor second to the iodine gamma monitor. Even for the case where the fission products are ejected from the fuel element by recoil, which favors delayed-neutron detection because the longer-lived Te and I activities will not be in equilibrium, the sensitivity of the iodine system will be orders of magnitude above that for delayed neutrons. The sensitivity ratio for a system using ion-exchange holdup will be even higher due to the buildup of the long-lived activities.

The delayed-neutron monitor offers, however, the distinct advantage of improved time resolution. In reactors having small-volume coolant loops this provides an improved method for the measurement of instantaneous fission-product production. In cores requiring large numbers of fuel assemblies the problem of finding the source of fission products requires the short time response afforded by this system.

mined, the gross iodine fraction can be analyzed to obtain the relative concentration of each isotope.

Once characteristic gamma spectra are available, it is possible to unscramble a pulse spectrum containing several different activities by successive subtraction of the different fractions from the gross spectrum. The usual procedures of gamma-ray scintillation spectrometry then yield quantitative ratios for the different isotopes present.

This provides a radiochemical method for determining the previous irradiation history of the source of fission products —information which is quite useful in localizing the source.

The iodine fraction is quite satisfactory for irradiation times up to several weeks. For longer periods the zirconium, rubidium, cerium, and strontium chemical fractions are used. The measurement of the isotopic ratios of 17-hr Zr^{97} to 65-day Zr^{95}, 4.5-hr Ru^{105} to 40-day Ru^{103}, 33-hr Ce^{143} to 32-day Ce^{141}, and 2.7-hr Sr^{92} to 58-day Sr^{89} provide age determination for prolonged periods of fuel irradiation. The use of isotopic ratios as opposed to ratios of activities of two chemical species removes any error that would result from selective processes that alter distribution of activities in the water system.

Iodine Chemistry

We have concluded that iodine activities found in the water stream can be considered representative of the quantity of gross fission products released to the water stream from exposed MTR fuel alloys. This conclusion followed a study of the behavior of iodine in the system.

Since the MTR water stream contains degassing equipment, the possibility that considerable quantities of iodine are removed at this point in the system had to be considered. Investigation showed, however, that very little iodine is removed from the system in the off-gas stream.

Studies of fission products deposited on the metal surfaces of the water system and scavenged by solid material carried in the water stream indicate that iodine itself is not removed from the stream although the tellurium parent activities are.

This would show that iodine that is formed by the decay of tellurium isotopes reenter the water stream very readily. The only remaining question —the possibility of selective evolution of fission products from exposed MTR fuel alloys—has been resolved to some extent. Specific chemical separations of barium, zirconium, molybdenum, and iodine were made from samples of process water containing fresh fission-product activity produced by exposed alloy fuel. Quantitative determinations were made for Ba^{140}-La^{140}, Zr^{95}, Mo^{99}, and I^{133}, observing chemical fractions with a scintillation spectrometer. No deviation was found from expected relative concentrations of these nuclides upon consideration of fission yield and irradiation time.

Observation of fission-product activities associated with ruptures occurring in high-temperature experimental facilities at the MTR indicates that iodine, being somewhat more volatile than many fission products, appears in much greater abundance than would be expected when it is compared with other fission products found. The information on this point to date is somewhat limited, and further studies are necessary. Our present feeling is that this assumption is not unreasonable and serves only to strengthen the case for radioiodine as a fission-product index in high-temperature air- and water-cooled systems.

Resin Studies

The properties of ion-exchange resins offer the necessary method for selective extractions from solutions. Conditions for their use are ideal in the water stream at the MTR. The *p*H is maintained within a narrow range (5–7), and the dissolved-solids content is generally below 1 ppm. Stable temperatures at sample points could be obtained in the range below 100° F. Cation and anion exchangers were prepared and studies made of the radioactive material collected from the coolant stream.

Cation resin. We used Amberlite IRA-120, a sulfonic-acid-type exchanger, as a cation resin. Samples were removed from the cation beds and examined with a gamma scintillation spectrometer to determine the nature of the material removed from the water stream. Comparisons were made between the distribution of radioactive species in samples of the water itself, the cation resin, and the resin effluent. Of the major background activities normally found in the water, Na^{24}, Cd^{115}, Ag^{110}, Zn^{65}, Co^{60}, and Fe^{59} were identified as major activities on the cation column. In addition, it became obvious that such a column acts as a filter for solid material as well as removing ions and charged colloidal particles.

Gamma analyses of cation-resin-effluent samples were made to establish the efficiency of the column for the removal of cation activity as well as to identify anions present. Figure 2 compares a gamma spectrum obtained from a liquid sample of the cation effluent taken from the MTR water system with that of an extracted iodine sample. In this case fission-product activity was present due to recoil of fission products from uranium deposited on the surfaces of the reactor components. The production rate from this source is constant and represents an equilibrium age distribution determined by the purge rate for the system. Activities identified include Mo^{99}, Te^{132}, and I^{132}, I^{134}, I^{135}. The delay between removal from the water stream and measurement prevents the observation of the short-lived gaseous fission products and their daughter activities. Kr^{87}, Kr^{88}, Xe^{133} and Xe^{135} have been observed in samples measured with minimum delay. Also plotted on the same figure is the pulse spectrum of the chemically separated iodine fraction taken under similar conditions. It is apparent that above 300 kev the effluent gamma spectrum is identical with that of the iodine chemical fraction.

Anion resin. The anion resin used was IRA-410, a strongly basic amine-type exchanger. Figure 3 summarizes the results of the anion-activity studies. This figure shows the decay of gamma emitters held on a resin sample removed from a column that operated in the MTR water system for several weeks under conditions of continuous fission-product production. The anion column was preceded by a cation exchanger and a filter to remove the solids as well as cations. Activities identified in the anion column included Te^{132}, Mo^{99}, Te^{99}, Cr^{51}, I^{131}, I^{133}, I^{134}, I^{135} and the xenon daughters of the iodine activities present. The decay of these activities was followed for several decades with the result that no activity with a longer life than 32-day Cr^{51} was identified. All of these nuclides with the exception of Cr^{51} are fission products.

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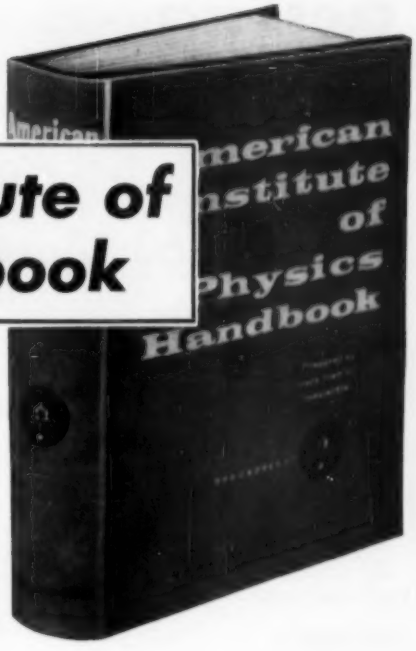

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Portable, Self-Powered Reader for DT-60 Glass Dosimeter

By JUSTIN C. SCHAFFERT
U. S. Naval Research Laboratory, Washington, D. C.

To read the dose on the U. S. Navy DT-60/PD phosphate-glass dosimeter,* it must be inserted into an auxiliary instrument (reader). The original readers were designed to operate on 115 volts, a-c, and weigh 25 lb. This paper describes the NRL Model B reader, a lightweight, compact, self-powered reader for field use.

Operation

The reader, shown in Fig. 1, operates continuously for 40 hr or more on one set of five size D, No. 2 conventional flashlight cells. The instrument measures $8\frac{1}{2} \times 8\frac{1}{2} \times 10$ in. and weighs 14 lb. It is feasible to redesign the reader with a size and weight saving of at least 50%.

The Model B contains the ultraviolet lamp and a filter to excite the fluorescence of the dosimeter, another filter and a photomultiplier to observe the fluorescence and an amplifier and meter-bridge circuit to indicate the dosage. The same mechanical and optical arrangement for accepting and positioning the dosimeter is used in the Model B as in the line-operated instrument.

The feature that permits extended operation on the small complement of flashlight cells is the use of a single short pulse of ultraviolet light from a Sylvania S-695 arc-discharge tube to excite fluorescence in the dosimeter. The S-695 is fired by two capacitors that are charged in parallel and discharged in series.

The photomultiplier provides a negative pulse that is proportional to the peak luminescent output from the dosimeter. The luminescent output is in turn proportional to the gamma exposure. The negative pulse is fed to a dual-stage transistor amplifier, which performs impedance matching and

FIG. 1. Self-powered DT-60 glass dosimeter reader. Index assembly is shown in CHANGE DT-60 position, in which dosimeter to be read can be inserted. With handle at CAL STD position, standard is in position to calibrate reader. At READ position, dose can be read

pulse inversion. The positive pulse from the amplifier is applied to a storage capacitor by means of a silicon diode. The capacitor charge is proportional to the peak pulse amplitude. Although the back resistance of the silicon diode is not infinite, the product of its back resistance and the capacitance is sufficiently large to permit retention of the charge on the capacitor allowing ample time for reading the meter-indicator.

The meter-indicator circuit is a vacuum-tube bridge composed of two subminiature vacuum tubes, the storage capacitor and a meter with scales of 0–200 and 0–600 roentgens. The meter indicates the amount of bridge unbalance due to the voltage across the capacitor. The meter may be "zeroed" by means of a zero control, which is used to balance the bridge circuit with the storage capacitor completely discharged. Coarse sensitivity control for the reader is provided by photomultiplier gain adjustment; fine control is provided by the calibration control. The range of the reader is set by the range switch.

Calibration of the roentgen scale of the meter is performed in the usual manner by reading a fluorescence standard and adjusting the reader gain to the appropriate value. The standard is a DT-60 dosimeter that was subjected to 3,000 roentgens from a cobalt-60 source. This dosimeter was then fitted with an optical mask to produce a reading of 150 roentgens on a Polaroid Corp. CP-95(XN-3)/PD reader. The masked dosimeter is permanently mounted in the index assembly. The index assembly is rotated by means of the handle shown in Fig. 1.

To calibrate the reader, turn the selector knob to ON. Set the meter to zero by means of the ZERO control. Place the sector handle in the CAL STD position and the range switch in the 0–200 roentgen position. After the READY light has fired, hold the TURN-TO-READ knob in its extreme clockwise position and note the

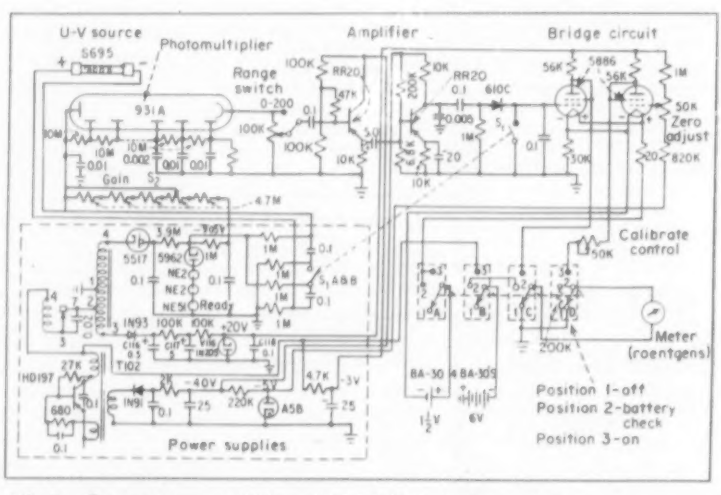

FIG. 2. Circuit diagram of NRL model B DT-60 dosimeter reader

* J. H. Schulman, W. Shurcliff, R. J. Ginther, F. H. Attix, NUCLEONICS **11**, No. 10, 52 (1953).

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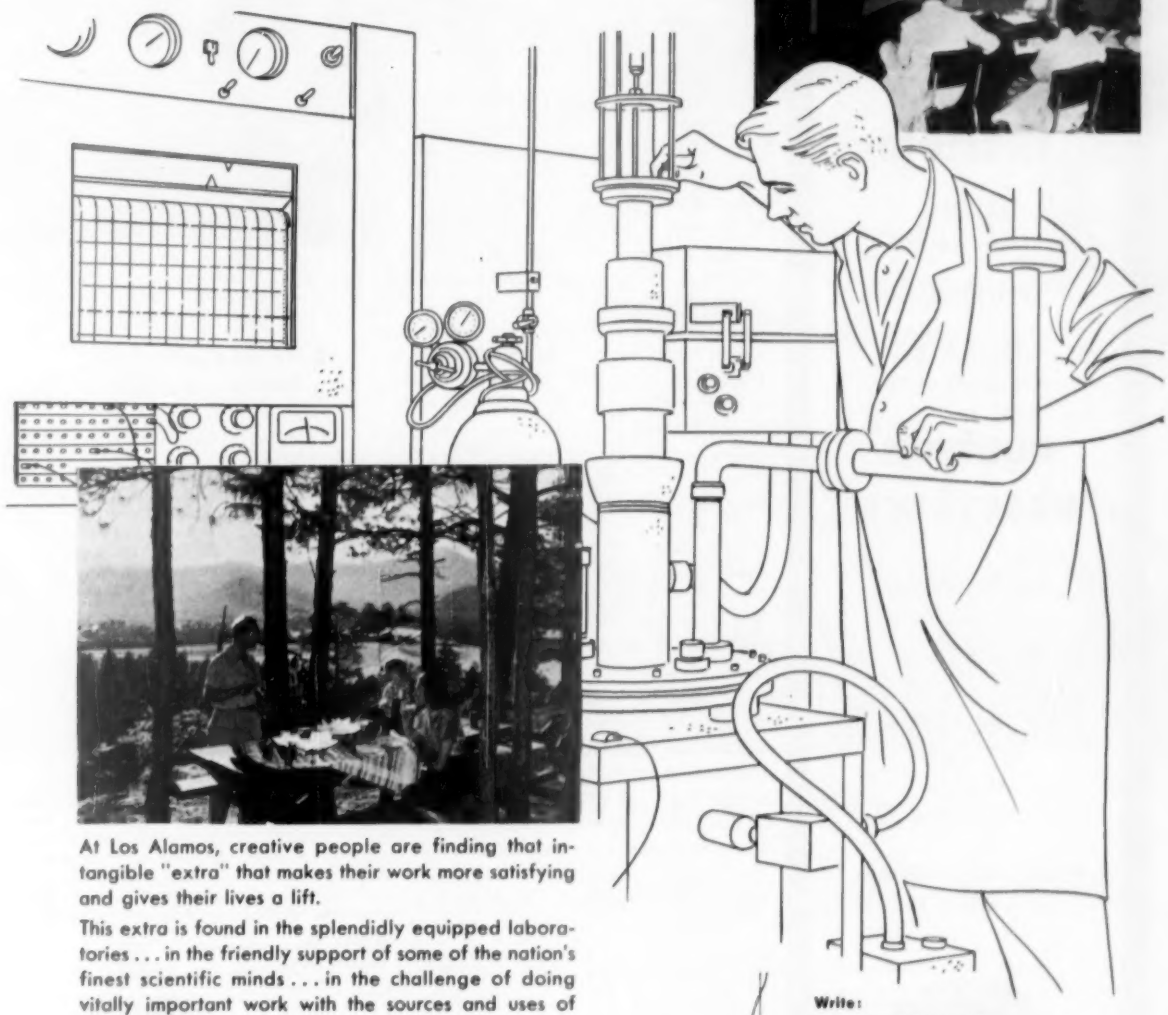

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meter reading. The calibration point is 150 roentgens on the 0–200 roentgen scale. While holding the TURN-TO-READ knob in its extreme clockwise position, adjust the CALIBRATE control so that the meter pointer coincides with 150 roentgens. If the pointer cannot be set to 150 roentgens by use of the CALIBRATE control, turn the GAIN control either clockwise or counterclockwise, depending on whether it is necessary to increase or decrease the meter reading, and repeat the procedure.

To read the dose on a DT-60 dosimeter, place the sector handle in the CHANGE DT-60 position and insert the DT-60 dosimeter. Now rotate the sector handle to the READ position. With the range set at the 0–200-roentgen position, hold the TURN-TO-READ knob in its extreme clockwise position. The meter indicates the dose directly. If the meter indication is off scale, release the TURN-TO-READ knob, set the range switch to 0–600 roentgens and repeat the operation. The READY light must be fired before each reading operation.

Repeatability

The chief source of spread in dose readings is apparently in the inherent variability of the discharge characteristics of the ultraviolet source. At present, little or no repeatability data are available on specific lamp types.

In the steady-state reader, wherein the lamp flashes are say 60 cps, this problem of repeatability is largely circumvented by using an indicator with a time constant that is long compared to the flashing period. Thus, the variation of the light output of successive flashes is "averaged out"; this is not the case for single-flash illumination.

The calculated standard deviation of fifty readings with a mean value of 145 roentgens was 8.1 roentgens. The probable error was 5.5 roentgens, or 3.75% of the mean value. The mean absolute error (arithmetic mean of all errors without regard to sign) was 6.5 roentgens or 4.4%.

The present spread of repeated dosimeter readings taken with the pulsed-light reader is not considered to be a serious deficiency in view of the overall accuracy of the DT-60 dosimeter system. However, in the interests of future reader designs, the repeatability of arc discharges should be studied experimentally with particular attention to the effects of aging of lamps used in flash service.

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Biological Response to Mixed Radiations

By DANIEL S. GROSCH, ROBERT L. SULLIVAN and LEO E. LaCHANCE
Marine Biological Laboratory, Woods Hole, Massachusetts and North Carolina State College, Raleigh, North Carolina

In reactor and certain accelerator situations radiation is often present as a mixture of radiations. To study the additive effects of such mixtures, we made a series of experiments on the combined sterilizing effects of external X-rays and ingested beta emitters (P^{32} and Sr^{89}) on the ectoparasitic wasp, *Habrobracon juglandis* (*1, 2*).

In all experiments, feedings followed the X-ray exposure. All combinations were comprised of doses that, by themselves have shown clear-cut effects. X-ray exposures were made with the Woods Hole M.B.L. generator (*3*) (2,500 r is one-half the sterilizing dose for X-rays). Phosphorus doses are listed in the table. Because of taste preferences of the wasps, strontium doses were limited to 500 μc/gm.

The graph summarizes the results. Except at the highest P^{32} level, the combination experiments more nearly resemble the X-ray-only results than an additive effect. Even at the highest P^{32} level, the influence of P^{32} on the combined effect is not proportional to the P^{32} effect alone. On the assumption that the X-ray and P^{32} effects are completely independent, the product of their survival percentages gives the theoretical values shown in the table. These are in close agreement with the experimental values. Chi-square values calculated from a 2×2 contingency tabulation supply no evidence against the hypothesis.

In contrast to radiophosphorus feedings, the Sr^{89} level does not induce permanent cessation of egg production (*4*). As shown in the graph, the values are consistently higher than the P^{32}-combination experiments and approach more closely the effects of X-rays alone. The less intense emission from the longer-lived Sr^{89} influenced egg production only slightly. Furthermore, unlike the P^{32} experiments, there is a recovery phase. On the eleventh day, eggs again began to appear and a day later a plateau was reached that held until senility (19–20 days). This recovery pattern does not differ from that following X-ray exposure alone (*2*).

With combinations of rays, the more intense component may do most of the damage. Presumably a majority of the sensitive cells were irreversibly damaged by X-rays and could not be further harmed by beta rays. Only a few cells, recovered from or never damaged by X-rays, were injured by beta rays.

Such independence of action supports two of the fundamental premises of theoretical radiobiology—additivity of two radiations may not be demonstrable unless (1) they have a similar

Sterilization Data for X-Rays + P^{32}

Expt.	*X-ray (r)*	*P^{32} mixture (μc/gm)*	*Day of onset of sterility*	*Egg survival (%)**		
				X-rays + P^{32}	*Theoretical X-rays + P^{32}†*	x^2‡
A	4,375	500	3	17	18.7	0.7270
B	3,750	500	3	20	22.7	1.0191
C	3,125	500	5	24.8	24.3	0.1809
D	2,500	500	6	31	35.3	1.3133
E	2,500	250	6	42.5	38.6	1.1285
F	2,500	750	6	27.5	26.9	0.1069
G	2,500	1,000	3	25.5	13.0	2.4003

* Obtained from (experimental eggs)/(control eggs).
† Based upon a hypothesis of complete independence of effect.
‡ Chi square is determined for 1 degree of freedom.

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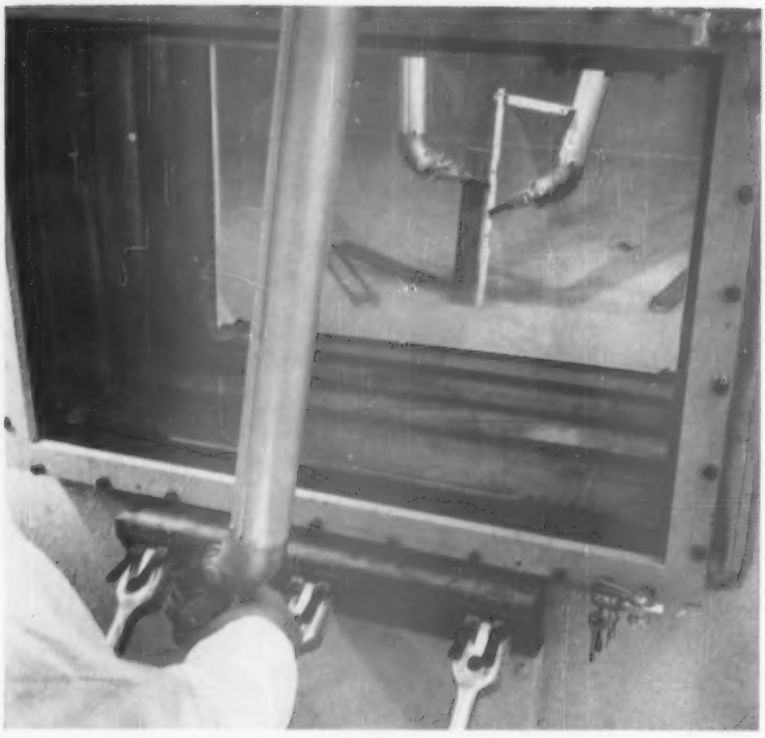

42-inch-thick window by Corning provides protection against cobalt 60. (Photo by Fritz Henle, courtesy Cities Service Co.)

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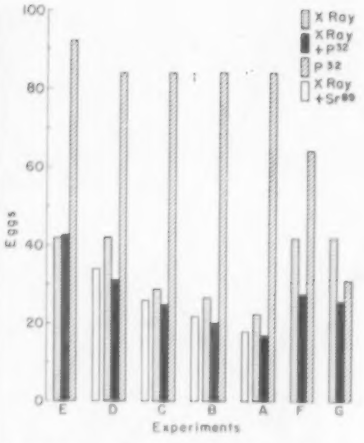

AVERAGE TOTAL OF EGGS laid per female during week subsequent to irradiation. Since control wasps produced slightly more than 100 eggs during same period, actual totals shown are essentially in per cent of controls. Alphabetical designation of experiments follows table

distribution in time and (2) their exposure times are brief (*5*). The latter condition, of course, cannot be met with fed radioisotopes but the present results should not be taken as evidence that the mechanisms of damage differ for X and beta rays. Lack of additivity merely indicates that the biological mechanisms influenced are in some way unrelated. In the present case the lack of relation may be entirely in time. Given time, developmental and restorative processes can proceed.

This suggests that in order to protect organisms from *immediate* deleterious consequences, we must particularly guard against sources of intense ionizing radiation. Here we are not considering recessive genetic damage wherein all ionizing radiation may be of concern.*

On the other hand, when differential damage is desirable, as in the elimination of pests from foods, mixtures of radiations can be quite inefficient.

* * *

These results are based on investigations performed under project AT-(40-1)-1314 for the U. S. Atomic Energy Commission.

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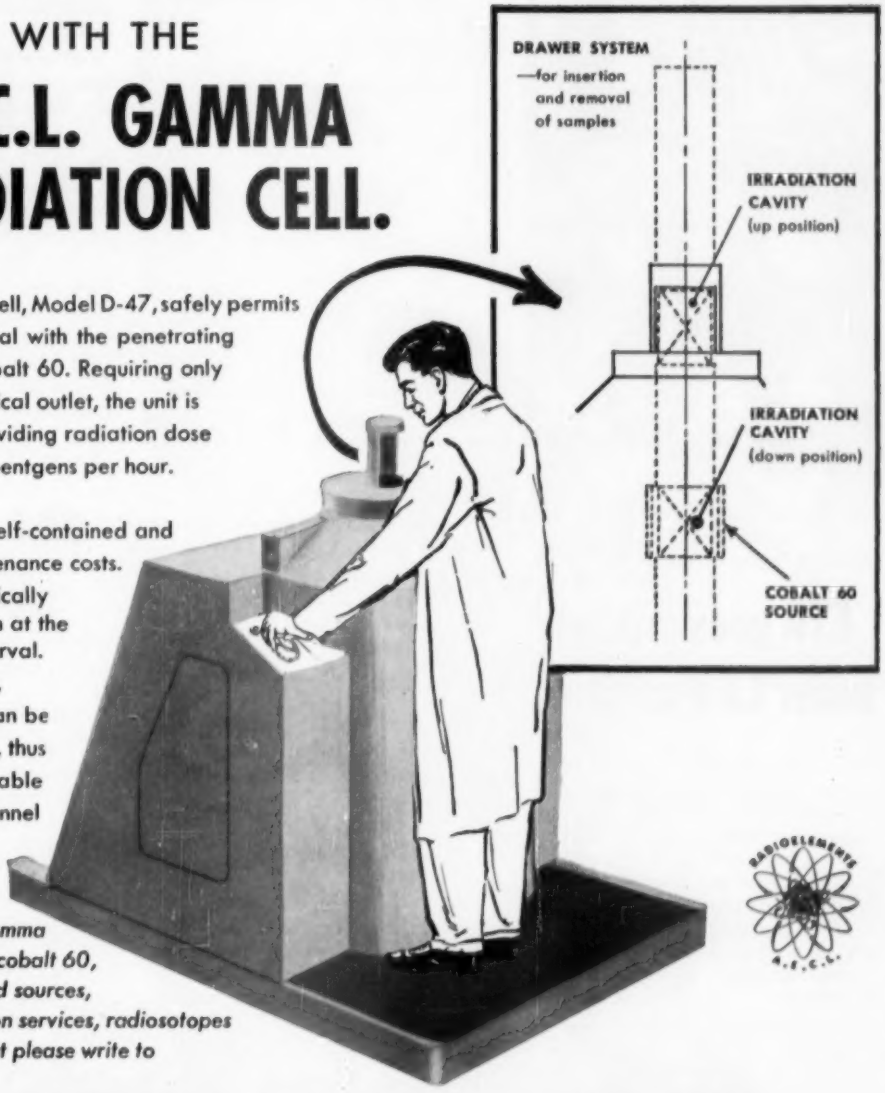

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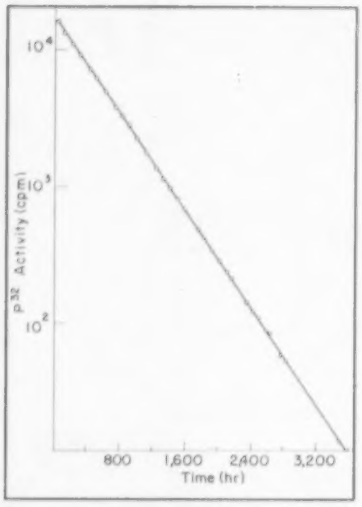
P^{32} Half-Life Determination

By OSWALD U. ANDERS* and W. WAYNE MEINKE
Department of Chemistry, University of Michigan, Ann Arbor, Michigan



DECAY of highly purified P^{32}. Sample was obtained from (d,α) reaction on sulfur

Phosphorus-32 is widely used as a biological tracer and as an intercomparison standard for beta counters. The recent publication (*1*) of a good radiochemical determination for this radioisotope promises its use as an analytical tracer.

P^{32} is particularly suitable as a tracer since it emits only one strong beta particle of 1.7-Mev maximum energy without admixed gamma radiations. It is readily available. Its relatively short half-life makes it safe and it is chemically easy to work with.

To use a standardized P^{32} solution for calibrating and intercomparing counters, it is essential that the half-life be known very accurately. This is necessary to correct for decay occurring since the standardization.

One of the first values reported was 14.30 ± 0.03 days (*2*). A value of 14.06 ± 0.07 was obtained from a source made by cyclotron bombardment of sulfur and measured with an electroscope and a G-M counter and corrected for a longer-lived impurity believed to be S^{35} (*3*). A source wrapped around a G-M tube and followed for two half-lives gave 14.35 ± 0.05 (*4*). A value of 14.60 ± 0.05 was obtained by following decay for three half-lives with an electroscope (*5*); the large value was probably due to lack of chemical purification. Calorimeter and G-M counter measurements gave 14.30 ± 0.09 (*6*); however, calorimeter determinations lack precision. The last available value was 14.50 ± 0.04 (*7*), obtained by following the decay for five half-lives with an electroscope.

In most of this work, the P^{32} samples were not pure, and corrections for S^{35} or P^{33} had to be applied. The P^{32} decay was followed for less than six half-lives, and the counting-geometry reproducibility and linearity of detection posed problems. These facts may easily account for the discrepancies reported and the errors quoted. To attempt to arrive at a significantly better value, we obtained a higher isotopic purity, provided a better reproducibility of counting conditions, and counted a larger number of half-lives.

A sample of high purity was obtained by producing the P^{32} by deuteron bombardment of pure ZnS. The 7.8-Mev deuterons employed were not energetic enough to produce any long-lived phosphorus isotope in addition to P^{32}.

The phosphorus activity was chemically purified in the following way: All cations were retained by passing the dissolved sample through a Dowex-50 bed in the hydrogen form. The P^{32} was then selectively adsorbed on a Dowex-50 $Fe(OH)_3$ column, in an adaptation of the method described by McIsaac and Voigt (*8*), and eluted with NaOH. The eluate was again passed through a Dowex-50 column, which retained all the sodium and yielded the P^{32} in carrier-free form.

The pure P^{32} was mounted on a 30-$\mu g/cm^2$ Zapon film, covered by a similar film and counted in a 4π beta proportional counter. This counter had a plateau of more than 800 volts with negligible slope and exhibited excellent linearity.

The decay of the sample was followed for 11 half-lives. A total of 66 decay points were taken during this time. The illustration shows a representative

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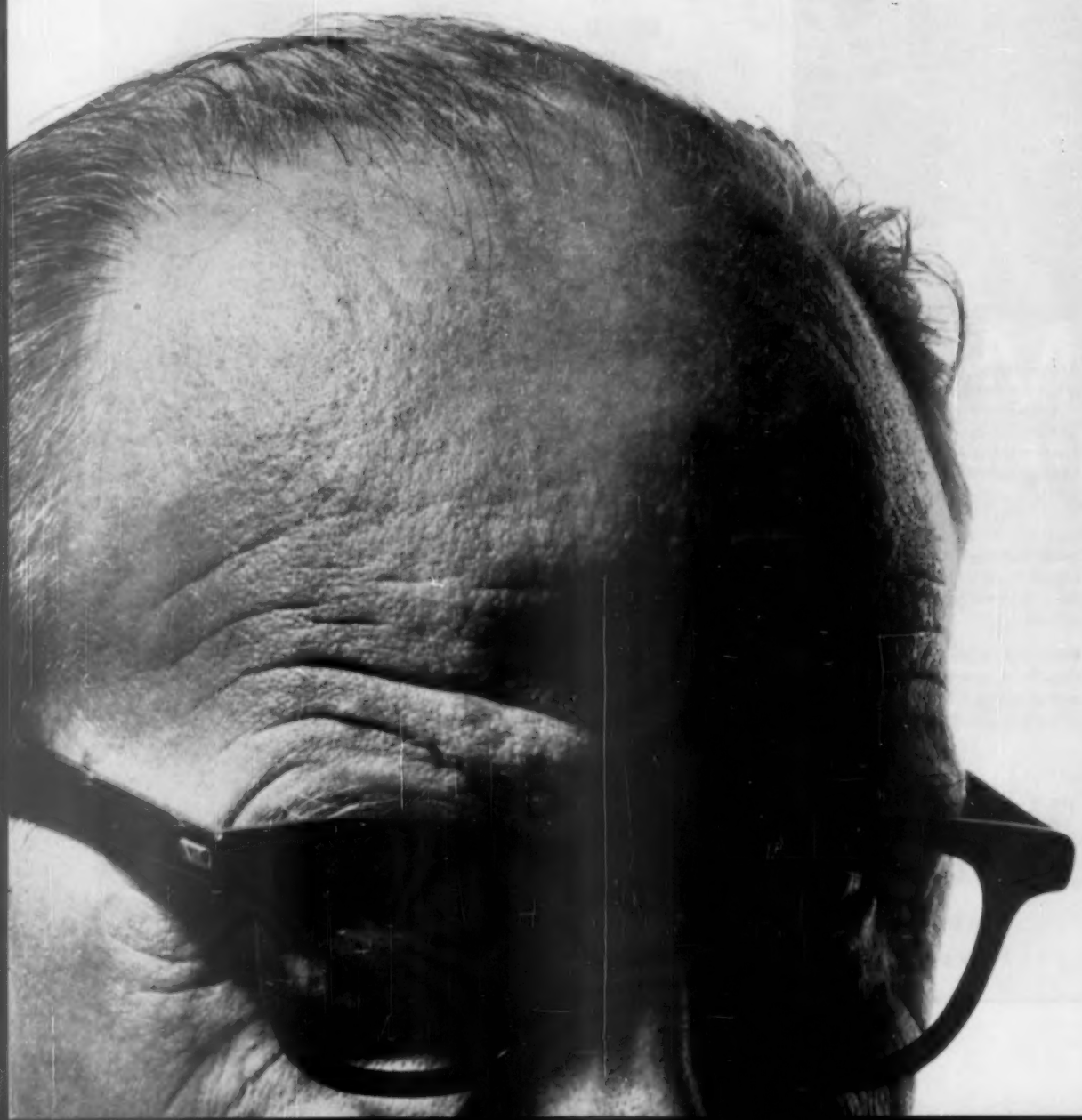

sampling of these points. Calculation by the method of least squares of the phosphorus-32 half-life from these points, corrected for background and coincidence losses, yielded a new value of 14.223 ± 0.015 days (standard deviation). Since this value is slightly smaller than most values reported, it would appear that systematic errors involving impurities and data scatter have been minimized in this experiment.

• • •

The authors are grateful to Mrs. Rosemary Maddock for her help in following the decay of the sample. The interest of Dr. W. J. Youden in the statistical treatment of the data is also appreciated. This work was supported in part by the U. S. Atomic Energy Commission.

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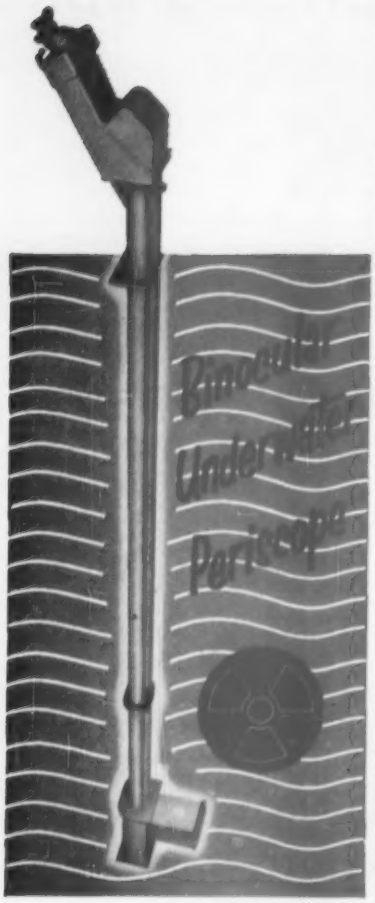

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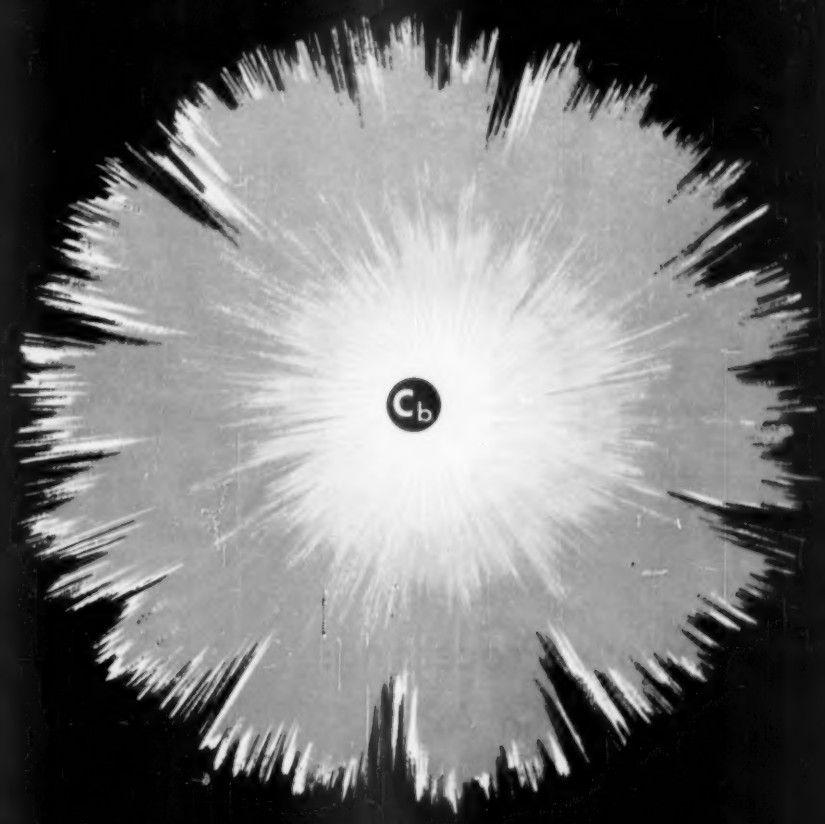
Hanford Adopts Plastic Film Badge

Slide showing shielding
Slide showing film packet in place
Body showing shielding and locking device
Assembled cover and body with slide partially inserted
Rear view of badge assembly
Badge assembly showing photo in place

New plastic personnel film badge dosimeter developed at Hanford has following advantages over old metal badge: lightweight construction (28 gm vs 74), reduced size ($5^{15}/_{32}$ in.^{2} vs $8^3/_8$), effective electromagnetically operated locking mechanism, simplified processing procedures (800/hr vs 350), improved dosimetry, and lower cost ($1.18 each vs $3.85).

New badge is made of relatively nonflammable cellulose acetate butyrate, which is available in any color. Three pieces of shielding include 1-mm-thick silver to absorb all beta and low-energy gamma radiation, 0.005-in.-thick silver to absorb gammas of below 15 kev, and 0.0191-in.-thick aluminum to absorb betas. Locking device consists of neoprene O-ring and cadmium-plated soft-iron "T". When slide is inserted into body, it forces "T" against O-ring until past protrusion. Strong electromagnet is required to pull "T" away from beneath slide protrusion so that slide can be removed. Soft-iron bar in bottom of slide allows it to be removed by second electromagnet—L. F. Kocher, General Electric Co., Hanford Atomic Products Operation, Richland, Wash.



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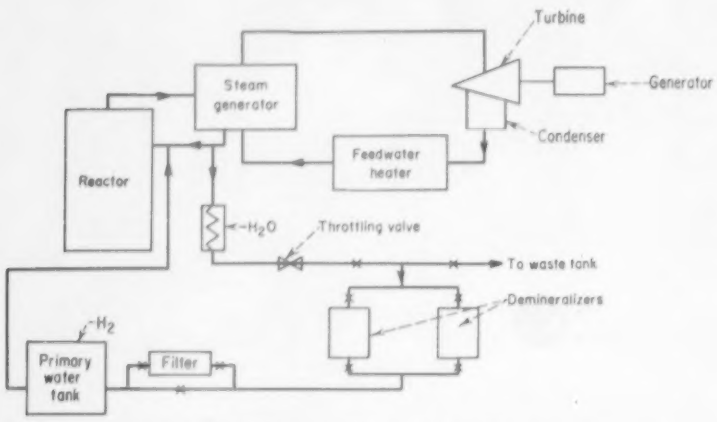


FLOW DIAGRAM of APPR showing bypass system for purifying primary water

Reactor Water Processing: How It Is Done in APPR

By A. LOUIS MEDIN
ALCO Products, Inc., Schenectady, N. Y.

In this era of nuclear power, many phases of plant operations formerly considered of secondary importance have acquired new significance. Water treatment, especially in pressurized-water reactors, is a good example. This article will describe the water-treatment facilities provided for the Army Package Power Reactor (APPR). Although the APPR is a relatively small pressurized-water reactor, the water-treatment problems of larger commercial power reactors should be similar.

The APPR (NU, Aug. '57, foldout) is a 10-Mw(th) system designed so that its major components are all transportable by air.

Water at 1,200 psia serves as both moderator and coolant and circulates through the enriched-uranium stainless-steel core at 4,000 gpm. As shown in the figure the water enters the reactor vessel at 431° F, is heated to 450° F, and then passes through a vertical U-tube steam generator. Steam at 200 psia is produced at a nominal temperature of 407° F, which represents about 25° F of superheat. The steam is used for driving the turbine, evaporator and air ejectors, plus some minor plant chores.

Essentially, all construction materials in the primary system are type 304 stainless steel. With the exception of the 304 stainless-steel steam generator, conventional construction materials are used in the secondary system.

The various water systems at the APPR are summarized in Table 1. The primary system includes all water and equipment in contact with the nuclear core, while the secondary system consists of all water and equipment in the steam-generating cycle.

Primary Water Treatment

Exceptionally high-purity water is required in the primary system of a water reactor for numerous reasons. Generally these reasons can be divided into two major categories—nuclear and mechanical. From the nuclear standpoint, impurities can pose radiation problems. If the impurities were to become radioactive and deposit in an area to which personnel have normal access, a serious radiation hazard could result. Impurities important in this respect include sodium, cobalt and manganese.

From the mechanical standpoint, impurities can result in the clogging of close-tolerance mechanisms. Deposits have caused serious malfunctions in the operation of control-rod mechanisms in at least one pressurized-water reactor.* In an operation as critical as the movement and positioning of control rods, continuous performance must be assured.

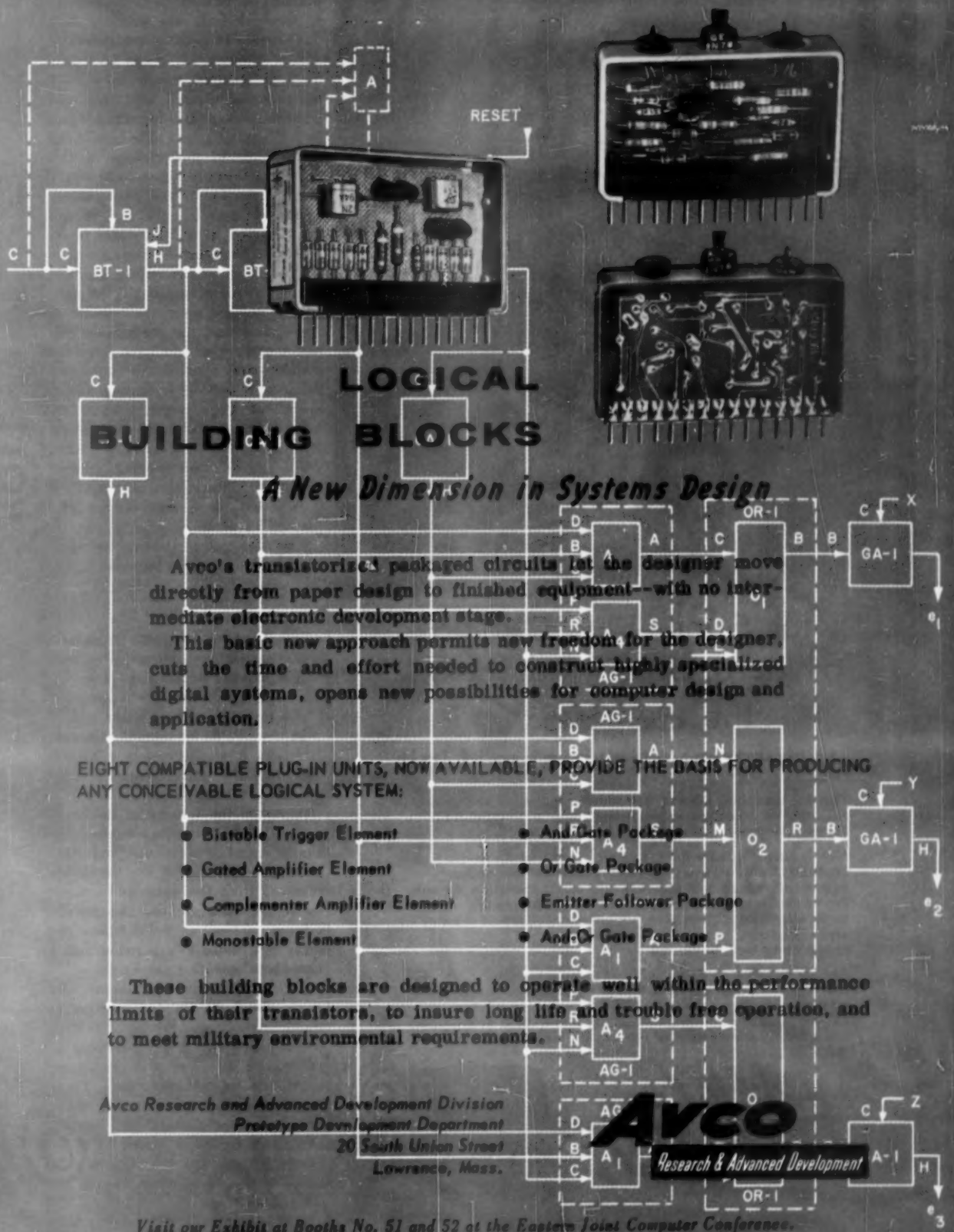
Deposition of impurities on fuel elements can also cause difficulties. Since the water gap between two adjoining elements is normally very small, deposits can create heat-transfer problems. Should the deposits become large, the normal water flow between elements could be disrupted.

In APPR, continuous purification of a portion of the primary water maintains the primary system at a maximum of 2 ppm total solids. Water requiring

* I. R. Welinsky, P. Cohen, J. J. Seaman, *Chem. Eng. Progr.* **53**, 393 (1956)

TABLE 1—APPR Water Systems

Type of system	*Use*
Service	Feed for evaporators
Distilled water	Makeup for secondary system, makeup for primary demineralizers, coolant for canned motor pumps, coolant for shield water
Secondary feed	Chemically treated water for feed to boiler
Primary	Demineralized water containing an excess of hydrogen
Spent-fuel pit	Distilled water under which fuel elements are stored
River water	Coolant for condenser, blow-down cooler and space coolers
Primary blowdown	Water exceeding specifications requiring purification and reuse
Seal water	Seals on control rod drives



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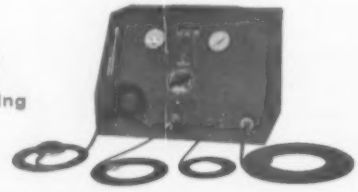

S. S. White Industrial Division, Dept. 9
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purification, referred to as blow-down, is removed from the primary loop downstream from the steam generator at approximately 431° F and 1,200 psia. The temperature is reduced to ~120° F by means of a small heat exchanger while the pressure is reduced to 100 psia by a throttling valve. The blow-down is then passed through one of two mixed-bed demineralizers where the bulk of the impurities are removed. Following collection in a 5,000-gal stainless-steel tank, the water is recirculated back to the primary loop.

The blow-down rate is a function of the corrosion rates, the primary system surface area, the volume of the system and the purification efficiency of the demineralizers. For the stainless-steel reactor, the blow-down rates will be higher during initial operations when corrosion rates are higher. After stabilization of corrosion rates, the blow-down will then vary as the demineralizer effluent changes. The corrosion rate in APPR is expected to average around 0.05 mg/cm²/month.

The blow-down rate can be calculated by the following formula

$$R = \frac{xA}{2{,}720(c - y)}$$

where R = blow-down rate in gal/hr
x = corrosion rate in mg/cm²/month
A = exposed surface area, cm²
c = purity of primary water, ppm
y = primary make-up water, ppm

In the APPR, for example, the surface area is approximately 4,000,000 cm² and the primary-water purity specification is 2 ppm.

Although the activity of the blow-down if diluted with the condenser discharge would be below accepted tolerances, the activity of the impurities removed from the water and concentrated in the demineralizer becomes significant. Therefore, the demineralizer must be shielded to prevent radiation hazards to operating personnel.

In conventional operations, the mixed-bed resins are regenerated after they have been exhausted. With resins containing large quantities of radioactive constituents, however, regeneration is not practical. There would be a problem of disposing of the large volumes of contaminated regenerating chemicals and water. The regeneration operation itself would be prohibitively complicated because of the radioactivity.

For these reasons the demineralizer units for the APPR were designed to be inexpensive enough so that both

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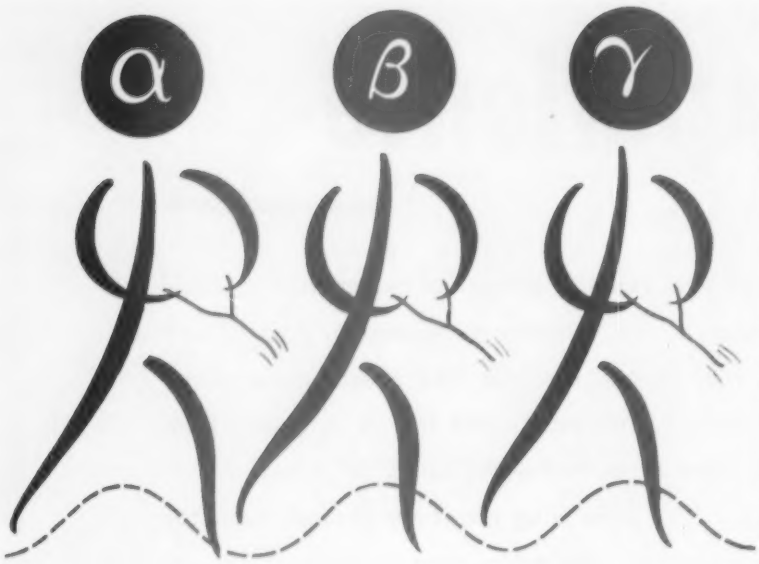

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TABLE 2—Demineralizer Activity

Nuclide	*Half-life*	*Activity (curies)**
Iron-59	46 days	0.35
Iron-55	2.9 yr	6.23
Chromium	26.5 days	53.4
Cobalt-60	5.2 yr	3.58

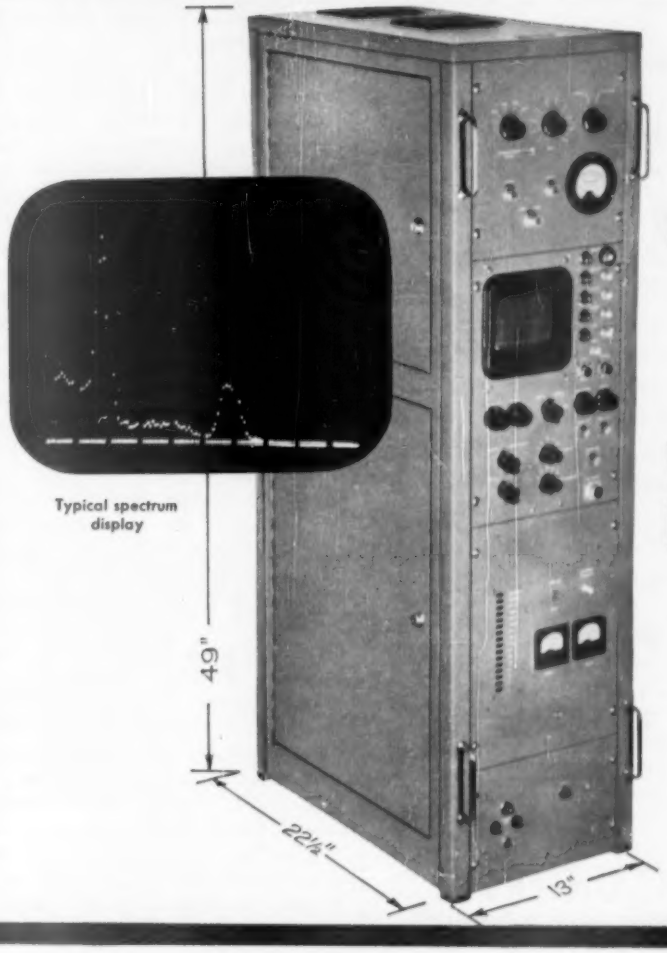
* The activities shown are for 24 hr after shutdown.

the resin and the container could be discarded. The unit is approximately 12 in. in diam. and 36 in. high; it contains about 2 ft³ of mixed resin. The construction material is type 302 stainless steel for the head and shell and 304 stainless for the tubing and fittings. The inlet and outlet pipes, which protrude from the top of the container, have been fitted with quick disconnecting fittings to permit rapid removal and assembly.

The corrosion products in APPR are primarily insoluble oxides of iron, nickel and chromium, although other impurities such as compounds of manganese and cobalt are also present. Normally, most of these products could be removed by filtration and resin life could thereby be extended. Unfortunately, most present-day filters become clogged easily and require frequent backwashing. To avoid the problem of handling backwash water containing radioactive nuclides, it was decided to use the demineralizer as both a filter and an ion-exchange unit.

A commercial ion-exchange material was sought that would produce good results and yet permit high operating temperatures. Unfortunately, no commercial mixed-bed resin can satisfactorily withstand temperatures above 140° F. The resin selected is a mixture of a sulfonic-acid-type cation exchanger in the hydrogen form and a strongly basic anion resin in the hydroxyl form. The resins are specially treated by the manufacturer to remove any soluble organic contaminants present. As received, the resins are in the regenerated hydrogen and hydroxyl forms, respectively, and have a minimum capacity of 12 kilograins/ft³ calculated as $CaCO_3$. This preparation by the manufacturer tends to guarantee a minimum leakage of water soluble material from the resins.

By estimating the probable activities that would be removed by the demineralizer, the units have been



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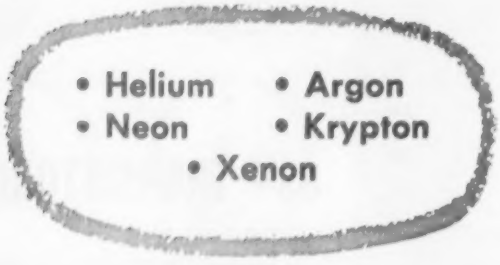

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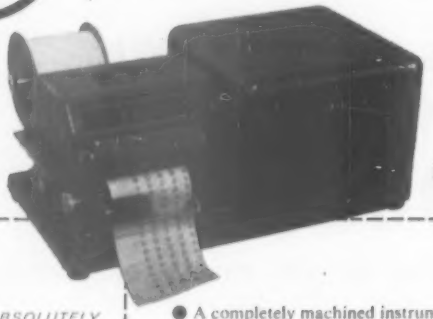
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sized to operate for approximately 3 months before disposal. The estimated activities are shown in Table 2. To meet ICC regulations, shipping the demineralizers to the disposal area requires about 4 in. of lead shielding.

To minimize corrosion in the primary system, oxygen is held as low as possible. Oxygen enters the system by in-leakage and from the dissociation reaction of water. While the reactor is in operation the hydrogen concentration is maintained at 15 to 45 cm^3/liter of water to minimize water dissociation. Hydrogen is admitted to the 5,000-gal make-up water tank to maintain a blanket of gas over the tank. During initial fill and startup when no radiation field is present, hydrazine is added to the primary water to scavenge oxygen.

A micrometallic filter is located downstream of the demineralizers to collect any resin leaving those units. The 304 stainless-steel filter elements will remove all particles larger than 7 microns. If during operations the filter does not appear necessary, the water can be by-passed around it.

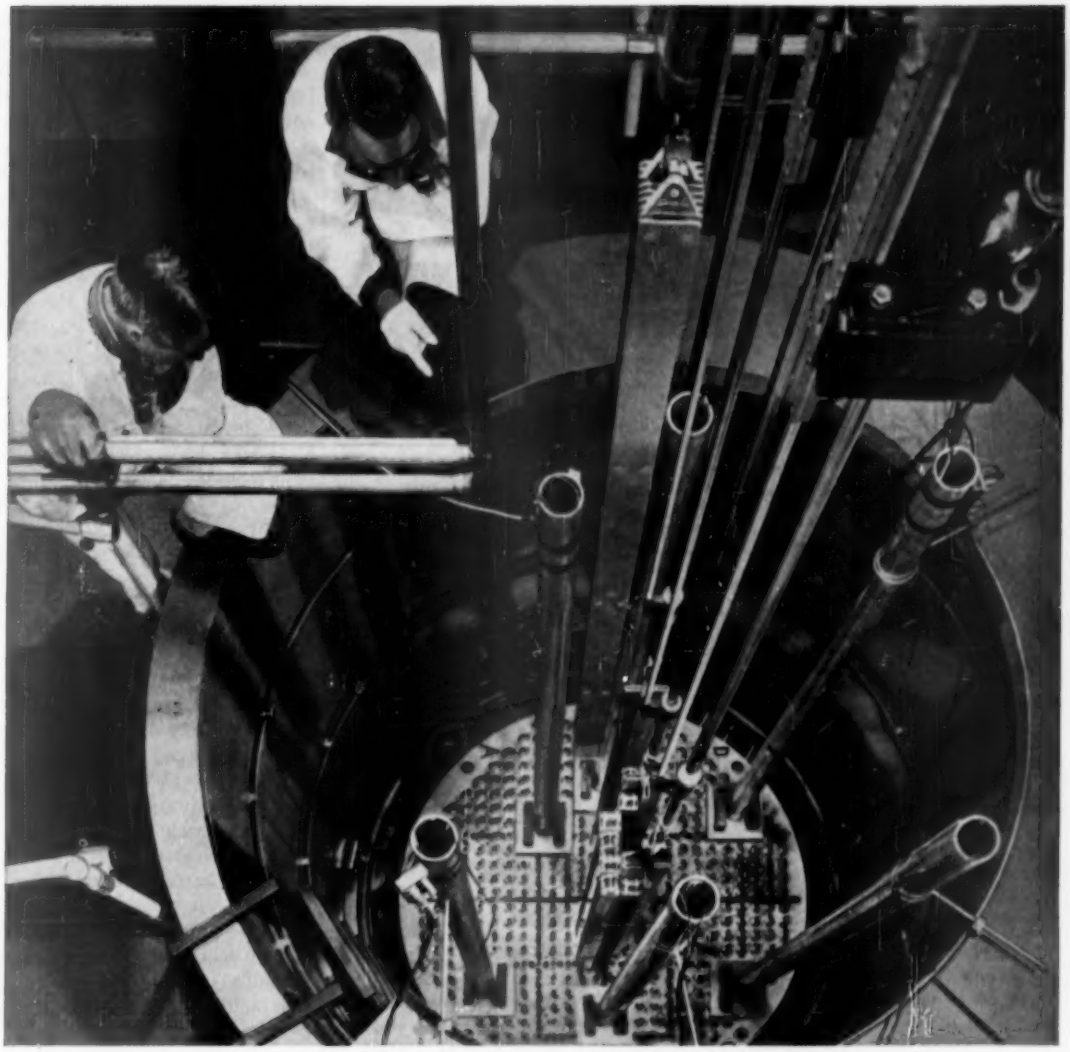
Control-rod-seal water is supplied by the primary make-up pumps. The water is collected in a seal leakage system, which maintains an excess of hydrogen to prevent air inleakage. Should additional make-up water be required for the primary system, deaerated condensate is used. The condensate is admitted to the primary system after passing through the mixed-bed demineralizer.

The possibility of a fuel element rupture appears extremely remote. Should, however, a fuel element cladding become damaged and fission products enter the primary water, provisions have been made to contain the wastes. If the primary blow-down water were to show excessive radioactivity, it would be automatically diverted to a 5,000-gal underground carbon-steel tank. Following collection of all the primary water, provisions would then be made to remove the damaged fuel element and dispose of the radioactive wastes.

Other than the two demineralizers and the filter, all other equipment in the primary blow-down or purification system has no shielding. This feature has not only permitted cost savings but has also minimized building size.

Secondary Water Treatment

Comparison of a conventional power plant's secondary system with that of



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a nuclear power plant shows them to be very similar. The main difference seems to be in the boiler. Whereas boiler-tube failures in a conventional power plant are certainly undesirable, in a nuclear plant the possible effect of such failures is much more serious. Should a leak or tube failure occur, radioactive primary water will enter the secondary system resulting in an accompanying radiation hazard to operating personnel. If the boiler is used to supply steam for building heat, the radiation problems become even more serious.

Compared with conventional power plant practice, therefore, the water-treatment requirements of the secondary water system are much more stringent. In particular, the possibility of chloride stress corrosion requires careful control of the chloride concentration. The necessity for chloride control imposes severe restrictions on the construction and operations of the secondary system. In APPR, the allowable chloride concentration has been set at 0.5 ppm. With a nominal blow-down rate of 1%, the feedwater must not exceed 0.005 ppm chloride if the 0.5-ppm limit is to be maintained.

After startup conditions have been achieved, very little makeup is required for the secondary system. The only make-up water required arises from the amount necessary to replace blow-down and unaccountable leakage. Careful control will be maintained during initial operations to prevent excessive buildup of chlorides in the boiler.

Make-up water will be prepared from wells having a hardness of 45 ppm as $CaCO_3$ and 85 ppm total dissolved solids of which 8.5 ppm are chlorides. This water will be used principally as feed for an evaporator. [The evaporator is a flex-tube (thermal descaling) type having an over-all capacity of approximately 2,150 lb/hr, of which 400 lb/hr will be used for the preheater deaerator.] The distillate from the evaporator will not exceed a total solids content of 1 ppm, of which the chloride content will approximate 10%. Since the makeup is only 1%, when the distillate is mixed with the condensate, the feed to the boiler will be below 0.005 ppm chloride.

Normal power plant practice provides both filtration and softening prior to evaporation. Such procedure is considered sound in ordinary power plant operation, but certain disadvantages arise for use in package power reactors to be located in remote facili-

ties. Although operation of either filters or softeners does not require extensive time or technical background, difficulties with their use are not uncommon. Along with the increased maintenance problems, the use of regenerative chemicals in remote facilities is to be avoided. Consequently, a thermal descaling evaporator was selected so that a substantial reduction in both equipment and operational problems is achieved.

The distilled water produced from the evaporator is stored in a 5,000-gal tank and is used for many purposes. The uses include, besides feed for the secondary system, (a) make-up water to the primary demineralizers, (b) coolant for the primary circulating pumps, (c) coolant for shield water, and (d) water for spent-fuel pit.

Chemicals used to control the secondary water composition follow ordinary power-plant practice. A deaerating hot well produces water containing 0.03 cm^3 of oxygen per liter, and to further reduce this amount, hydrazine is fed to the boiler feed. Should the addition of hydrazine not prove feasible, sodium sulfite can also be added. To scavenge CO_2 a morpholine is added to maintain the pH of the condensate return at approximately 8.5–8.9.

About 16,000 gal of water are contained in the spent-fuel pit. The water is circulated periodically through a diatomaceous earth filter to prevent it from becoming dirty, cloudy or contaminated with algae.

Chemical Analyses

During initial startup and precritical operations, extensive sampling and analysis is required on both the primary and secondary systems and normal analysis is required on the other water systems.

Other than periodic radiochemical analysis, Table 3 shows the constituents that are analyzed by APPR along with the normal range of limits.

The established methods were found inadequate to meet many of the lower desired limits. This being true especially for the gas analyses, new methods had to be developed. An example of a typical developed method is that for hydrogen. Hydrogen is measured by means of a standard Orsat technique, which provides sufficient accuracy in the desired range of gas concentrations. At atmospheric pressure where the gas analysis is conducted, much of the hydrogen will remain in solution so that it must be stripped from the water before satisfactory measurements can be performed. The equipment developed for this purpose eliminated the use of high vacuum apparatus.

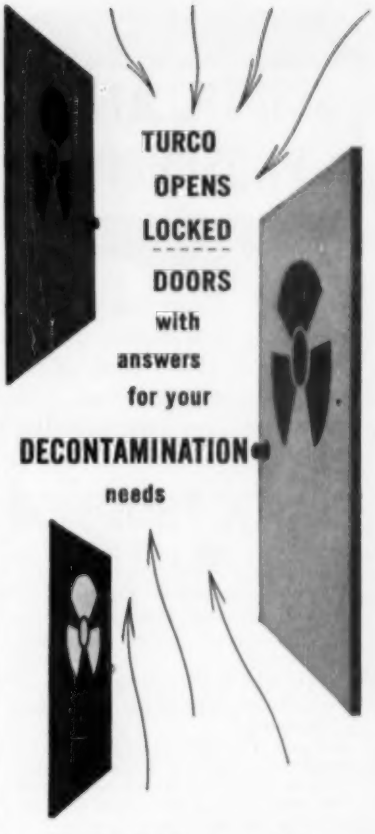
The method is based upon a liquid-vapor extraction column operating in a fixed volume at reduced pressure. The hydrogen and other noncondensable gases are stripped from the liquid. The gas extraction is possible as a result of the reduced solubility of hydrogen at the lowered partial pressure, which is accomplished by dilution with air and the maintenance of low total pressure. The hydrogen partial pressure at the end of the extraction is about 10 mm and after compression to 100 ml is about 40 mm. At these pressures, extraction is about 99.6% complete.

TABLE 3—APPR Chemical Analyses

Analysis	*Location*	*Normal limits*
Chlorides	Primary, secondary	0–1.0 ppm
Oxygen	Primary, secondary	0–1.0 cm^3/liter
Hydrogen	Primary	0–30 cm^3/liter
Total solid	Primary	1–2 ppm
Total solid	Secondary	0–1,000 ppm
Total dissolved solid	Primary	1–2 ppm
Total dissolved solid	Secondary	0–1,000 ppm
Hydrazine	Primary, secondary	0–0.5 ppm
Iron	Primary, secondary	0–1.0 ppm
Cobalt	Primary	0–0.1 ppm
Nickel	Primary	0–0.5 ppm
Manganese	Primary	0–0.5 ppm
Chromium	Primary	0–0.5 ppm
*p*H	Primary, secondary	
Conductivity	Primary, secondary	
Phosphate	Secondary	0–10 ppm
Sulfite	Secondary	0–10 ppm
Hydroxyl ion	Secondary	0–10 ppm
Ammonia	Primary	0–0.5 ppm

* * *

The author wishes to acknowledge the many suggestions and ideas offered by Dr. J. L. Meem, former chief reactor scientist for ALCO Products, Inc. and now a consultant to the company, in the design and development of the APPR water treatment systems. Dr. Meem is now professor of nuclear engineering at the University of Virginia.



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Rod Drives for Water Reactors*

A major component of most reactors is the control rod drive. In water-moderated and -cooled reactors such as pressurized-water and boiling reactors, the drives must perform satisfactorily under various temperature and pressure conditions in an environment of hot water or steam. The main problems in designing these drives are corrosion, sealing, and fast insertion (scram).

The corrosion problem is normally solved by the use of such corrosion resistant materials as stainless steels (types 304, 316, 347, and 17-4PH), for many of the component parts such as shafts, guide rollers, pressure tubes, etc.; stellite, for bushings in linear bearings and the balls in roller bearings; and Inconel or Cobenium, for spring materials.

The sealing and fast insertion or scram problems normally go together. Under a scram condition it is usually preferable that absorber safety rods be inserted quickly into the reactor under the force of gravity although they may be assisted by other forces due to hydraulic cylinders or preloaded springs. To accomplish this, the inertia and friction of the drive system, which remains attached to the safety rod, must be kept as small as possible. This requires that the rod work through low-friction seals or without seals and that as much of the drive system as possible be disconnected from the rod. A brief review follows on some of the types of rod drives that are currently being used or being tested for use in water reactors.

1. Canned Motor Drive

The drives for the submarine Nautilus and for the Shippingport Reactor, PWR, are of the canned motor type (*1, 2*). In the PWR, which is operated at a pressure of 2,000 psi and 542° F, the rods are enclosed in thimbles on top of the reactor pressure shell. The seal problem is solved by using a canned motor wherein the rotor is separated from the stator by a nonmagnetic cylindrical can capable of withstanding the reactor design pressure. The prototype mechanism uses corrosion resistant materials throughout. The latch and scram mechanism consists of a 2-arm collapsible mechanism, actuated electromagnetically, in which the drive nuts are disengaged from the lead screw during a scram allowing the rod and lead screw assembly to fall free under the force of gravity and a preloaded accelerating spring. The rod is damped at the end of its stroke by a water dash pot inside the pressure vessel. There being no seals, this system has no normal leakage and friction is very low.

A second drive mechanism (*3*) is being studied as a possible replacement for the original mechanism. It consists of a fixed roller nut and a lead screw that is continuously engaged to the motor rotor. On a scram the lead screw must drive through the roller-nut assembly under the force of gravity assisted initially by a spring-operated, fluid-coupled kicker mechanism. The scram speed of this type of mechanism is substantially slower than the prototype discussed previously. In hot autoclave tests, this mechanism operated without a failure for 5,624 cycles and 2,006 scrams.

In both of these drives the canned motor temperature is kept at approximately 200° F by cooling coils built into the motor. The main disadvantage of this system is the high cost of the canned motor drive.

2. Rotary Seal Drive

a. The Army Power Package Reactor (APPR) is a pressurized water reactor operating at 1,200 psi and 450° F (*4*). It makes use of rack and pinion drives mounted on the bottom of the reactor pressure vessel. The pinion is driven by a commercial gear motor through a labyrinth type, low friction, high pressure rotary seal. Since the make-up water for the primary system is introduced on the high pressure side of the seal, the small amount of leakage through the seal is largely uncontaminated water that is processed along with the primary-system blowdown water. The materials used are again the 316 or 347 stainless steels and stellite balls for the rotary bearings. An external electromagnetic clutch connects the gear motor to the drive. When this clutch is de-energized, the rod is

*Excerpted from "Power Reactor Technology—A Summary of Recent Developments" prepared by Dr. Walter H. Zinn and associates of the General Nuclear Engineering Corp. under AEC contract.

allowed to scram under gravity, rotating the pinion assembly while falling. A water dashpot is used to damp the rod at the end of the stroke.

This reactor has been operating intermittently since its dedication without any trouble having been reported on the rotary-seal system.

b. Another reactor, the Argonne Low Power Reactor (*5*, *6*), also is designed to use a rack and pinion drive working through a rotary seal. This drive is mounted on top of the pressure vessel. The seal is of the labyrinth type, which is kept cool by introducing cooling water on the high pressure side of the seal and collecting the leakage on the low pressure side. During a scram, a magnetic clutch is de-energized, disconnecting the pinion shaft from the drive system and allowing the safety rod to drop under gravity assisted by a preloaded accelerating spring.

3. Linear Seal Drive

a. The Experimental Boiling Water Reactor (*7*) is a boiling water reactor that operates at 600 psi and 489° F. Control rod drives for the reactor are mounted on the bottom of the reactor and the rods are positioned by motor driven screw and nut assemblies. The nut is integral with a carriage, which carries the control rod extension rod and is electromagnetically coupled to the screw. The chrome-plated extension rod passes through a labyrinth type (pressure breakdown), high pressure, low friction linear seal. A nylon guide bushing is used to align the extension rod as it passes out from the reactor through the seal.

It was originally intended that the seal would be kept cool (110° F) by introducing a small amount of feedwater to the rod thimble equal to the leakage from the seal. However, in operating the reactor, it was found that this cooling water caused a drop in the lower-vessel head temperature, which could be troublesome from the thermal-shock hazard (*8*). Since the average leakage rate for these seals is very small, and since the operating temperature, without cooling and at normal reactor power, of the thimbles was only 150° F, the forced-cooling-water system will probably be eliminated and an intermittant leakage makeup system substituted.

The rods are scrammed by de-energizing the carriage nut-latch magnet, freeing the rod from the screw, and allowing the reactor pressure, as well as an accelerating spring, to drive the

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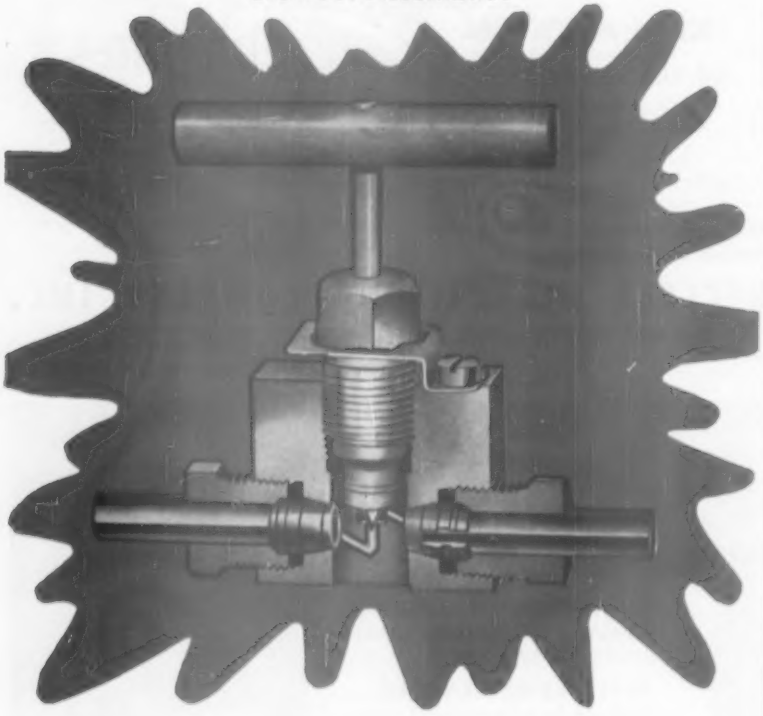

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extension rod through the linear seal; the friction of this seal is low enough that the rod will scram with gravity only. An external hydraulic dashpot is used to damp the rod at the end of its stroke. A second internal water dashpot is incorporated inside the pressure vessel to prevent damage to the control rod in case of failure of the external dashpot.

Experience to date (*8*) with this drive has been satisfactory except for some difficulty with the nylon guide bushing above the seal. Iron dust accumulated in the 0.0017-in. radial clearance of this bushing causing the rod to bind at zero reactor pressure. This trouble was eliminated by increasing the radial clearance to 0.0076 in.

b. The Dresden Nuclear Power Station (*9, 10*), which is a boiling water reactor designed to operate at a pressure of 1,000 psi and temperature of 544° F, makes use of hydraulically driven control rod drives mounted on the bottom of the pressure vessel. The hydraulic fluid is water, which is integral with the water in the reactor, and the piston operates through a packed linear seal. Positioning of the rod is determined by a motor-driven stop against which the rod is held by maintaining pressure on the up side of the piston. Scramming of the reactor is accomplished by removing the normal driving pressure and pressurizing the down side of the pistons by hydraulic accumulators.

4. Magnetic Jack Drive

A different approach to safety-rod drive systems is the use of linear-type electric motors of the incremental motion type. Many such linear motors have been experimented with, but most have been unsatisfactory in operation. Recently ANL has revealed a magnetic-jack type of electromagnetic drive, which shows some excellent promise as a control rod drive for water reactors. Since this was described in a recent issue of NUCLEONICS (*11*), it will not be described further here.

* * *

Full text available from Industrial Information Branch, Technical Information Service, U. S. Atomic Energy Commission, Washington 25, D. C.

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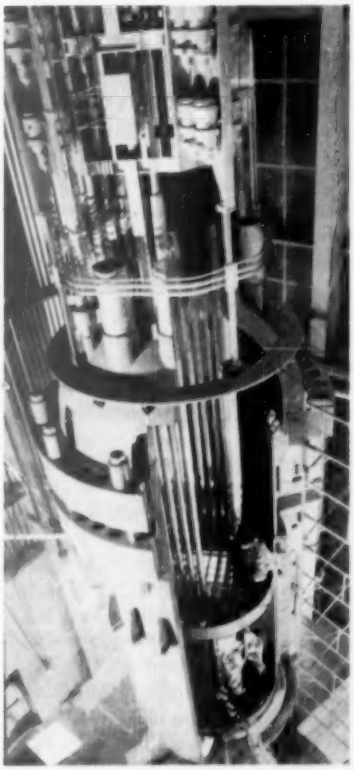
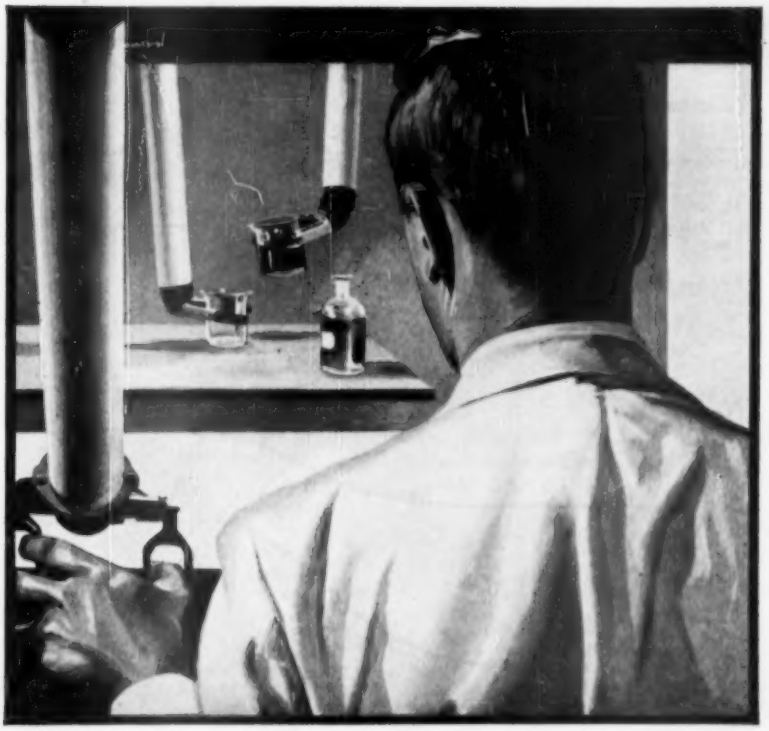
PWR-Core Mockup



Actual-size mockup of PWR 45 ft high and 10 ft dia is used by Westinghouse engineers at Bettis to determine shape and configuration of pipes, cables and components to be installed at Shippingport. Built of wood, plastic and metal, model is strong enough to handle eight tons of material in its structure. One side is open to permit engineers to enter. At top are seen instrumentation devices for checking operation; below (larger open pipes) are refueling ports. Three of 42 bolts fastening pressure vessel's head are shown; each is almost 7 ft long (cf NU, Dec. '56, 26). Long tubes ("shrouds") to left of man crouching on upper grid plate will contain control rods; to left of man standing on lower grid plate are movable dummy fuel elements.



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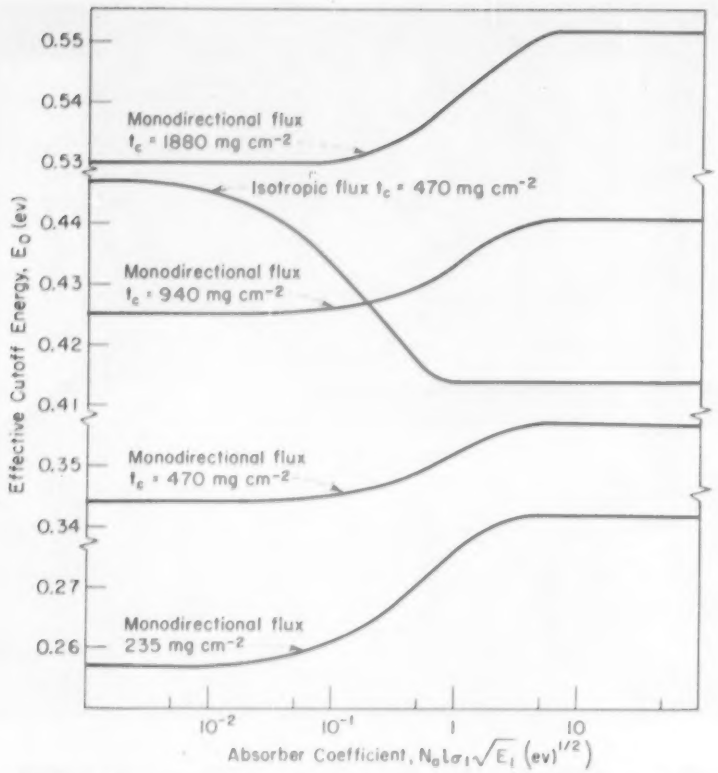

FIG. 1. Effective cutoff energy as a function of absorber coefficient for various cadmium thicknesses

Effective Cadmium Cutoff Energy

By I. E. DAYTON* and W. G. PETTUS
Atomic Energy Division
The Babcock & Wilcox Co.
Lynchburg, Virginia

CADMIUM is commonly used in neutron transmission and activation experiments as an absorber for thermal neutrons. Its high thermal cross section (about 2,500 barns) combined with an absorption resonance at 0.178 ev produces an excellent high-pass filter (*1*). The effective cutoff energy of cadmium filters is generally taken to be between 0.4 and 0.5 ev (*2*). We derive here more precise values (see Figs. 1 and 2).

Effective Cutoff Energy

To arrive at a precise definition of cutoff energy, consider first a perfect filter—a filter that absorbs all neutrons of energy less than $E = E_0$ and transmits all neutrons above this energy. If the scattering cross section of the sample is small compared to the absorption cross section, the number of absorptions per unit area of the sample will be proportional to

$$N_a \int_{E_0}^{E_{max}} \iint \varphi(E,\mu)\sigma_a(E) e^{-\frac{N_a \sigma_a(E) z}{\mu}} \frac{d\mu}{\mu} dz dE \quad (1)$$

where $\mu = \cos\theta$ (θ is the angle between the neutron path and the normal to the surface), z is distance into the sample measured along the normal to the surface, $\sigma_a(E)$ is the absorption cross section of the sample, N_a is the atomic density of the sample and $\varphi(E,\mu)$ stands for incident neutron flux. The limits of integration for μ and z are chosen appropriate to the geometry at hand.

When a cadmium filter is substituted for the perfect filter, the low-energy end of the flux spectrum incident on the sample is distorted as shown schematically in Fig. 3. If E_{th} = energy of thermal neutrons, $\sigma_c(E)$ = absorption cross section of cadmium, N_c = number of cadmium atoms per unit volume and t_c = thickness of cadmium filter, then the number of absorptions per unit area in a sample behind a cadmium filter will be proportional to

$$N_a \int_{E_{th}}^{E_{max}} \iint \varphi(E,\mu) e^{-\frac{N_c \sigma_c(E) t_c}{\mu}} \sigma_a(E) e^{-\frac{N_a \sigma_a(E) z}{\mu}} \frac{d\mu}{\mu} dz dE \quad (2)$$

The effective cutoff energy for a cadmium filter is defined as the cutoff energy of a perfect filter that allows the same total number of absorptions in a sample as the given cadmium filter. In other words, the effective cutoff energy is that value of E_0 which makes Eq. 1 equal to Eq. 2.*

It follows that the effective cutoff energy is a function of cadmium thickness, neutron flux spectrum and angular distribution, and sample material, size and shape. At first it appears fruitless to define the effective cutoff energy in a way that makes it a function of so many other variables, however, it turns out that the preceding definition is actually very useful. The problem that most frequently arises is the determination of fractional activation or self-shielding in a thick sample behind a cadmium filter. By using the above definition of effective cutoff energy, the problem can be separated into two parts:

1. Determining the fractional activation in a thick sample behind a perfect filter (*3*), and
2. Determining the effective cutoff energy of the cadmium filter that gives the same activation.

Application to $1/v$ Absorber

Cadmium filters are primarily used when irradiating samples in a nuclear

* PRESENT ADDRESS: Dept. of Physics, Swarthmore College, Swarthmore, Pa.

* Although the effective cutoff energy as defined here is not the same as the energy at which one-half the incident neutrons are transmitted, the "50%-transmission" energy can be used as a rough approximation.

reactor or in the beam from a reactor. The neutrons are slowing down from fission energies (about 2 Mev on the average) to thermal energies (about 0.025 ev) and the spectrum is given by, or is assumed to be given by, $\varphi(E)dE \propto dE/E$. For irradiations taking place in the interior of the reactor the angular distribution is assumed to be isotropic. Beam irradiations are characterized by a unique angle of incidence.

With the exception of some cases to be mentioned later, the concept of effective cutoff energy is most needed when dealing with $1/v$ absorbers. It is convenient to represent the absorption cross section by $\sigma_a(E) = \frac{\sigma_1 E_1^{1/2}}{E^{1/2}}$, where σ_1 is the cross section at energy E_1. Substituting the above expressions into Eqs. 1 and 2 and specializing to the case of isotropic flux incident on a slab of thickness l gives the following equation to be solved for E_0

$$\int_{E_{th}}^{E_{max}} \int_0^1 \int_0^l e^{-\frac{N\sigma_c(E)t_0}{\mu}} e^{-\frac{N\sigma_1 E_1^{1/2} z}{\mu E^{1/2}}} E^{-3/2}\, dz d\mu\, dE = \int_{E_0}^{E_{max}} \int_0^1 \int_0^l e^{-\frac{N\sigma_1 E_1^{1/2} z}{\mu E^{1/2}}} E^{-3/2}\, dz d\mu\, dE \quad (3)$$

In the numerical work that follows, E_{max} was taken as 10 Mev, but in analytical work infinity could be used without introducing any error. A similar expression for beam irradiations can be obtained from Eq. 3 by setting $\mu = 1$ and eliminating the integration over μ. This entails no loss of generality, since other values of μ merely correspond to cadmium and sample thicknesses of t_c/μ and l/μ, respectively.

In solving Eq. 3, it is convenient to introduce an analytical expression for $\sigma_c(E)$. In the work reported here $\sigma_c(E)$ was taken equal to the cadmium absorption cross section $\sigma_{ca}(E)$ which is given by the Breit-Wigner one-level formula

$$\sigma_{ca}(E) = \frac{a}{(E)^{1/2}[(E - E_r)^2 + b]}$$

(The experimental data are fitted to within about 2% in the energy region from thermal to 1 ev with the following constants: $a = 10.4$, $b = 3.22 \times 10^{-3}$, and $E_r = 0.178$ ev.) This is equivalent to assuming that a scattering does not remove an incident neutron. A more rigorous treatment would involve integrating over the scattering angular distribution, but, since in the neighborhood of 0.4 ev the scattering cross section is only a few percent of the absorption cross section (*1*), scattering effects can be neglected.

The integration of Eq. 3 with respect to z can easily be carried out in closed form. The remaining double integration was performed numerically on a digital computer, using twenty equal intervals in μ and 200 equal intervals in $\log (E_{max}/E)$.

Results of the calculations are plotted in Figs. 1 and 2. The range of parameters includes values ordinarily encountered in experimental work. Values of E_0 for other combinations of cadmium thickness and sample thickness can be had by interpolating between the curves.

The results for the beam geometry show, as one might expect, that as the sample thickness increases, the neutron spectrum hardens and the value of E_0 rises. No such simple physical explanation can be given for the shape of the isotropic curve.

The foregoing analysis can also be applied directly to materials with absorption resonances, provided the resonances are located far above the effective cutoff energy. It is only necessary

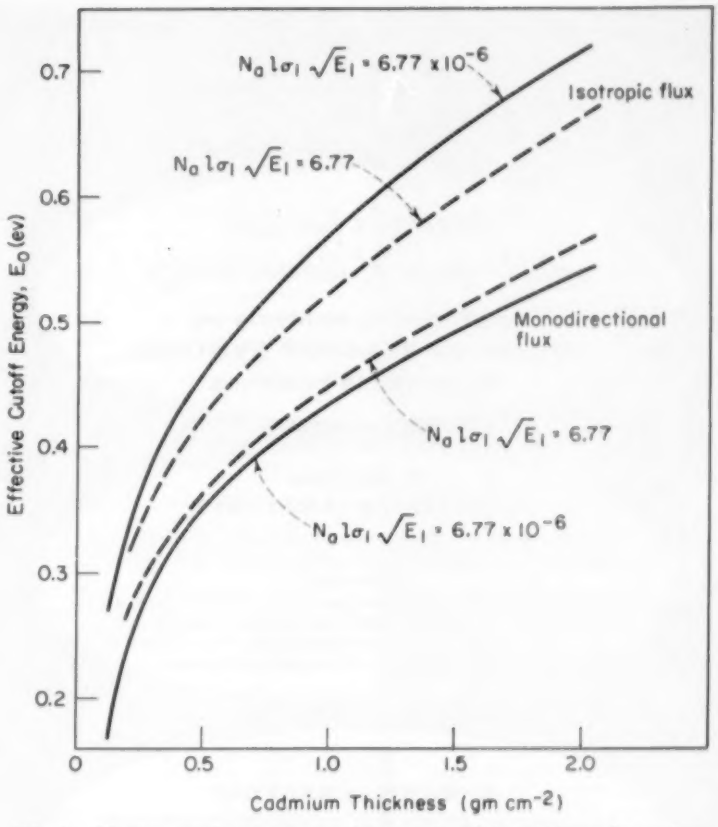

FIG. 2. Effective cutoff energy as a function of cadmium thickness for various absorber coefficients

How to Use These Curves

These curves can be used to find the effective cutoff energy for a cadmium neutron filter shielding a $1/v$ absorber in slab geometry. The cutoff energy E_0 can be obtained from either curve by specifying the cadmium thickness t_{cd} in mg/cm^2 and calculating the adsorber coefficient, $N_a l \sigma_1(E_1)^{1/2}$. N_a is the nuclear density of the absorber, l is the absorber thickness, and σ_1 is the microscopic absorption cross section of the absorber at neutron energy E_1. E_0 is given for both isotropic and monodirectional neutron fluxes. In each case the flux is assumed to vary as $1/E$ and scattering in the absorber has been neglected.

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to consider the $1/v$ component of the cross-section and proceed as above.

There are, however, a number of important materials—for example, In^{115}, U^{235} and Pu^{239}—that have resonances near the cadmium effective cutoff energy. The results presented here will not be applicable to these cases, but

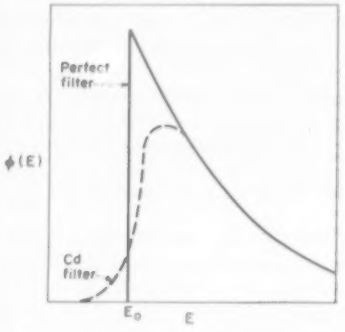

FIG. 3. Neutron transmission as a function of energy through perfect filter and through cadmium filter

the same method can be followed if the correct expressions for $\sigma_a(E)$ is used. For In^{115} the problem has been approached experimentally by determining the fraction of the In^{115} absorption resonance at 1.44 ev shadowed by the tail of the cadmium cross section (*4*).

* * *

It is a pleasure to thank Mr. Richard F. Reiss for programming this problem for the computer.

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Official Report on Windscale Accident (Continued from p. 43)

been known for some time but when the Windscale pile was designed and built knowledge was scanty. That such energy could be released spontaneously or by annealing was not clear at that time.

7. A spontaneous release of Wigner energy occurred in the Windscale No. 1 pile in September, 1952, whilst the pile was shut down. This led to a rise of temperature of the graphite but the rise was not dangerous and there were no harmful effects. As a result of a study of this incident, a procedure was instituted for controlling release of Wigner energy; and eight such releases had been carried out by the end of 1956 in Pile No. 1. The general procedure was to shut down the pile, arrange the appropriate instrumentation and then cause the pile to diverge with no coolant airflow, thus raising the uranium and graphite temperatures and so starting the Wigner release in the graphite. The release is then self-sustaining.

8. It has always been found difficult to release energy in all the graphite of the pile. Consequently, on three previous occasions, it was found necessary to apply a second nuclear heating, some time after the first nuclear heating, when it was seen from the behaviour of the thermocouples in the graphite that the release of Wigner energy was coming to a halt before all the graphite had been annealed.

9. On the present occasion, the pile was shut down at 01.13 on 7th October, and the main blowers switched off in preparation for the Wigner release. All necessary steps were taken to verify that the pile was completely shut down. The thermocouples were checked and those which were unserviceable were replaced.

10. The pile was first made divergent for the Wigner release at 19.25 on 7th October, and the nuclear heating was stopped early the following morning. Some hours later, it appeared to the operators then in charge of the Wigner release operation that the graphite temperatures were dropping rather than rising, and it was therefore decided to apply further nuclear heating.

11. The Committee of Inquiry carefully examined the temperature records on this point. Undoubtedly some of the graphite temperatures were falling and there were some in which no Wigner release was apparent; but a substantial number of graphite thermocouple readings showed steady increases. The Committee would not, therefore endorse the observation that the general tendency was for the graphite temperatures to be dropping rather than rising.

12. However, acting on the observations which he had made, the Physicist in charge decided to boost the release with a second nuclear heating. He had no Pile Operating Manual, with special sections on Wigner release, to help him, nor had he had the benefit of sufficiently detailed instructions.

13. The pile diverged for the second nuclear heating at 11.05 on Tuesday, 8th October. The uranium thermocouples showed a temperature increase lasting for about 15 minutes after the pile diverged. The Committee of Inquiry studied the temperature-time records and decided that the maximum rate of rise of temperature for a few minutes was several times greater than that which is permitted in normal operations. However, the maximum uranium temperature *recorded* did not exceed the maximum normal operating temperature.

14. The rapid rate of rise of the uranium temperatures was seen and the control rods were run in to reduce the pile power and thus allow the uranium fuel to cool somewhat. However, in the opinion of the Committee of Inquiry, the damage leading to the accident had probably been done. One or more of the uranium cartridges in the lower front part of the pile had failed.

15. There were two separate and distinct faults of instrumentation affecting the second nuclear heating. First, the uranium thermocouples were situated at the region of highest uranium temperatures during normal operations and were not at the position of maximum uranium temperatures during Wigner release. Because of this, the pile operator was unaware that the uranium fuel in the middle front regions of the pile was probably hotter than was indicated by the readings he was taking, and the rate of rise of temperature was certainly greater than he was observing. A second deficiency of instrumentation was that the pile power meter, whilst correct for normal operations, read low in the situation prevailing. The "feel" of the pile was different and the second nuclear heating was applied too quickly.

16. The picture that the Committee pieced together revealed that, as a result of the second nuclear heating, graphite temperatures gradually rose through Wednesday, 9th October. This led to the oxidation of the uranium which had been exposed by the overheating. The exposed uranium smouldered throughout the course of the day, 9th October, and gradually led to the failure of other cartridges and their combustion, and to the combustion of graphite. By Thursday evening the fire had spread and was affecting about 150 channels in a region of rectangular cross section.

Action Taken

17. During Wednesday, 9th October, the graphite temperatures showed considerable variation but the general tendency was for the temperatures to increase following the second nuclear heating. One graphite temperature in particular continued to rise steadily. By 22.00, the temperature had risen to such a value that the pile Physicist in charge was required, by a written instruction which he had, to have certain operational measures taken. These measures were taken as follows. The chimney base and the inspection holes had been shut at 21.00. The effect was to cause the chimney to draw a little air through the pile and thus cool it, rather than to draw air through the chimney base and the inspection holes. At 22.15, the fan dampers were opened for 15 minutes to give positive air flow through the pile. The dampers were opened again for 10 minutes at 00.01 on the 10th October, for 13 minutes at 02.15 and for 30 minutes at 05.10. These damper openings had a cooling effect on all graphite temperatures except the highest one, where the rise was merely arrested.

18. At 05.40 on the 10th October, at about the end of the fourth damper opening,

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Official Report on Windscale Accident *(Continued from p. 89)*

the pile stack activity meter near the filter at the top of the stack showed a sharp increase. This was noted by the Physicist who was then on duty but no special action was taken because he regarded it as a normal consequence of first movement of air through the pile and up the stack.

19. Between 05.40 and 08.10 the pile stack meter showed falling readings, but at 08.10 the activity began to rise steadily. Soon after noon, high activity was reported on the roof of the Meteorological Station.

20. During the morning of 10th October, the graphite temperature started to rise again, after the temporary halt caused by the damper openings, as explained in paragraph 17. At 12.10 therefore, the dampers were opened for 15 minutes, and at 13.40 were opened for 5 minutes. During the openings, the pile stack meter showed a sharp increase.

21. These effects suggested to the operating staff the existence of one or more burst cartridges. At 13.45, the shut down fans were switched on as a necessary preliminary to an attempt either to use the scanning gear to detect any burst cartridges or, alternatively, to blow the pile cool. At 14.30 the turbo-exhauster was switched on in order to scan for the burst slug. It was then found that the scanning gear was jammed and could not be moved. The Pile Manager stated that at the end of previous Wigner releases the scanning gear could not be moved, presumably because of overheating. On the present occasion it was immovable despite the fact that the Maintenance Section had worked on it the previous day, 9th October, and had moved it.

22. Being unable to operate the scanning gear, recourse was had to another activity-measuring instrument which could take samples of air coming from the pile provided that the turbo-exhausters were running. This meter showed a large reading.

23. At this stage, the Works General Manager was informed by the Pile Manager that there appeared to be a bad burst and he instructed that the affected channel should be identified and discharged as soon as possible.

24. Since the scanning gear could not be moved, it was decided to remove a charge plug in the charge wall in front of the pile, and inspect the channel showing the highest temperatures. Certain steps had to be taken before entry could be effected. An air count had to be taken, to ensure that there were safe working conditions; and the operators themselves had to have a change into protective clothing. Soon after access had been secured to the charge hoist, and the hoist raised to the correct height, the charge plug in the pile wall giving a view of the four uranium channels including the one where the thermocouple reading was highest, was removed. The uranium cartridges were seen to be at red heat.

25. The first move was an attempt to discharge the glowing channels but they were found to be wedged owing to the distortion of the cartridges; and prolonged efforts continuing until the following morning were only partially successful in clearing these channels. The next move was to create a fire break by discharging channels adjacent to those which were red hot. This succeeded in containing the area of the fire for some hours but did nothing to reduce the temperature of the affected area.

26. During the night, an attempt was made to bring down the temperatures by use of CO_2 brought over from the Calder Hall site, but this proved ineffective because by then the affected region was too hot.

27. Meanwhile, thought had been given to the possible use of water and by about midnight the decision was reached that if other measures continued to prove ineffective, water should be used. This was the major decision and it was supported by the Director of Production and the Deputy Works General Manager. The Works General Manager, the Director of Production and the Deputy Works General Manager all had in mind that although the fire break appeared to be containing the fire to the 150 affected channels, there was the possibility that further Wigner releases might occur in other regions of the pile, and add to the total heat in the pile.

28. Soon after midnight, the Chief Constable of Cumberland was warned of the possibility of an emergency and men in the factory were warned of an emergency with instructions to stay indoors and wear face masks.

29. By 03.44 the water hoses were ready to be coupled at 15 minutes' notice. At 07.00 it was decided that water should be used but that before it was turned on, all factory labour should be under cover. Some delay was therefore necessary while the shift changed over at 08.00. Water was first turned on at 08.55 but without immediate effect. About an hour later the shut down fans, which had been kept in operation in order to maintain tolerable working conditions on the charge hoist (where the unloading operations were being attempted), were closed down. This operation, combined with the continued use of water, caused the fire to subside. Pouring of water continued for over 24 hr until, by Saturday afternoon, the pile was cold.

30. The Committee of Inquiry advised the Atomic Energy Authority that in their opinion the steps taken to deal with the accident once it had been discovered were prompt and efficient and displayed great devotion to duty on the part of all concerned.

How the Committee Determined the Cause of the Accident

31. The method by which the Committee of Inquiry reached their conclusion that the second nuclear heating probably caused the failure of one or more uranium fuel cartridges, which then smouldered and gradually caused the fire, is highly technical. A number of calculations were made for the Committee, and these, together with the temperature-time records and the known sources of heat, enabled the Committee to determine the maximum temperatures in the pile, and the flow of heat from one part to another. The rate of rise of temperature of the uranium cartridges in the lower front part of the pile, where the trouble started, was found; and the maximum temperature reached by these cartridges, during the second nuclear heating, was estimated.

32. The Committee of Inquiry considered evidence on the behaviour of fuel cartridges in experimental furnaces. They also considered operational data on the failure rate of fuel cartridges in the pile, and the way in which the failure rate depended on the time that a fuel element had been in the pile.

33. The uranium fuel cartridges in the lower front part of the pile, amounting to many hundreds of cartridges in all, had been there for some time. The uranium rods, therefore, had grown somewhat, by radiation damage, and some of the cans were in a state of strain. If the cartridges had been raised slowly in temperature only to the normal maximum operating temperature, by means of a nuclear heating, it is most unlikely that any would have failed. However, these fuel cartridges were raised well above the maximum normal operating temperature; and the maximum rate of rise of temperature was several times the rate of rise in normal operations. Whilst the argument is one of probability rather than certainty, the Committee of Inquiry considered that the most likely cause of the accident was the combined effect of the rapid heating and the high temperature reached by the fuel elements in the lower front part of the pile. In all probability, one or more end caps of the cans of fuel elements were pushed off, and uranium exposed.

34. The Committee carefully examined all the scientific records taken in order to be sure that it was possible to reconcile their view that uranium cartridges failed during the second nuclear heating, at about 11.15 on 8th October, and smouldered without any of the instruments recording that something untoward had occurred. The Committee are satisfied that the scientific records taken during Tuesday, Wednesday and Thursday are consistent with their explanation of the accident.

35. The Committee made a detailed study of cartridges containing lithium-magnesium alloy, a number of which were in the pile for the purpose of irradiation. After evaluating the highest temperatures which any of these cartridges could have reached as a result of Wigner releases following the second nuclear heating and after studying the evidence concerning temperatures at which such cartridges have been observed to fail, they came to the conclusion that the cause of the accident could conceivably have been the failure of a lithium-magnesium cartridge but that this was unlikely.

36. The pile also contained a number of other cartridges where various isotopes for industrial and civil research were being formed. The Committee examined each of these in turn and rejected the possibility that any of these cartridges was the cause of the accident.

37. Attention was also given to the possibility that, irrespective of the second nuclear heating, a large local release of Wigner energy occurred in a pocket of graphite which had not been annealed for some time, and that as a result high local temperatures were caused, sufficient to result in the failure of a fuel cartridge, or a lithium-magnesium cartridge, or even the combustion of graphite. The Committee of Inquiry studied the thermocouple records, laboratory data on the failure of cartridges, and general information. The Committee rejected explanations of this type.

38. There are vertical channels in the pile used for experiments connected with the civil reactor programme. At the time of the accident, all of these channels were empty except one which contained a small magnet under test. There were no experiments either for civil or military purposes being done in the pile during the time of the Wigner release, except for the magnet mentioned in the previous sentence.

Note by the U. K. AEA on the Calder Hall and Civil Power Reactors

1. The accident which happened at Windscale could not happen with the Calder Hall reactors or with the reactors being built for the Electricity Authorities. Important differences exist between the Windscale Piles, the more recently designed Calder Hall reactors, and the still more advanced reactors for the Electricity Authorities. The later reactors incorporate improvements based on the experience obtained in designing and operating the earlier reactors.

2. In the Calder Hall and Electricity Authorities' reactors Wigner releases will be required much less frequently. The minimum temperature of the graphite at Calder Hall in normal operations is higher than the corresponding temperature at Windscale. The minimum operating temperature of the graphite in the first stations now under construction for the Electricity Authorities will be higher still. For this reason some annealing of the graphite will be going on continuously in the Calder Hall reactors and in the civil power reactors, and the storage of Wigner energy in the graphite will proceed at a much slower rate. All the evidence accumulated up to the present time indicates that these reactors will be able to operate for about five years before a Wigner release is required. This is being continuously checked by monitoring of the graphite in Calder Hall and by accelerated tests of graphite in the DIDO research reactor.

3. When a Wigner release becomes necessary, it will not present any hazard because it can be conducted under fully controlled conditions, viz.:—

(a) Whereas the Windscale Piles are cooled by air the Calder Hall type reactors are cooled by carbon dioxide which does not react with uranium until appreciably higher temperatures than those at which the air reaction starts. In still air uranium oxidises, i.e. the reaction is self-heating at 350° C; the corresponding temperature with carbon dioxide is 650° C–700° C. The safety margin during a Wigner release operation will, therefore, be very much greater.

(b) In the Windscale Piles the air

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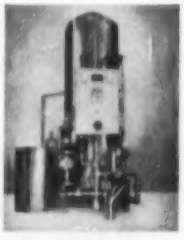

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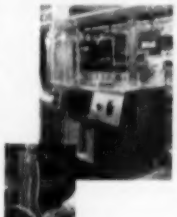

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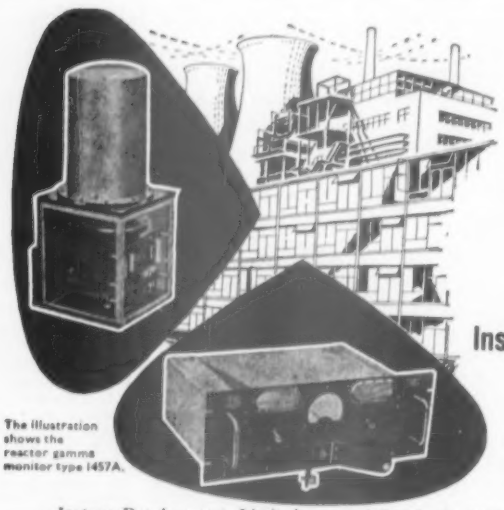

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coolant is discharged into the atmosphere through a high stack. In the Calder Hall reactors the carbon dioxide coolant circulates in a closed circuit. The use of carbon dioxide as a coolant in a closed system makes it possible to keep the temperature generated by a Wigner release under close control by the use of the main cooling fans.

(c) The scanning gear in the Windscale Piles is of the moving type and is designed to move upwards and downwards across the Pile discharge face. At the high temperature experienced during the Windscale accident this gear jammed. The detection equipment in Calder Hall and in the Electricity Authorities' stations is fixed and provides a method of monitoring each fuel element channel without movement. Continuous monitoring, therefore, of the integrity of the fuel sheath will be carried out while Wigner releases are in progress, thus providing continuous information about the condition of the fuel elements. This will enable immediate remedial action to be taken to deal with any damaged fuel element.

(d) The fuel cartridges in Calder Hall are of much improved design and experience has shown that their failure rate is much lower than the failure rate of the Windscale fuel elements, and the rate of attack by carbon dioxide has been shown to be very slow. The fuel cartridges for the Electricity Authorities reactors will incorporate these and further improvements.

(e) The instrumentation and control system for regulating, inter alia, the rate of pile power change in the Calder Hall and Electricity Authorities' reactors is much improved over those in the Windscale reactors. These improvements will provide additional important safeguards.

4. As stated above the graphite in Calder Hall is being periodically checked for storage of Wigner energy and accelerated tests of stored energy are being carried out in the Harwell research reactor DIDO. If these and other tests should show any new factors, there is ample time to introduce additional safeguards, e.g. by filling the reactor circuit with an inert gas during the Wigner releases.

5. Even if a fuel cartridge failed in a reactor of the Calder Hall type and a reaction began between the uranium and the coolant causing a release of fission products into the coolant, it would be immediately detected for the reasons given above and could be brought quickly under control by using the main fans to reduce the temperature. The amount of radioactive material which could be released into the closed circuit would, therefore, be small, and the amount that could escape into the atmosphere from leakages in the closed circuit would be too small to constitute any hazard.

APPLIED RADIATION

New Dimensions in Autoradiography

Filter paper has been used to extend autoradiography in plant studies. Instead of simply exposing a photographic plate to the radioactivity in a leaf, for example, you squeeze the leaf on filter paper and then make an autoradiograph of the paper. Several techniques are available to make the result specific for different substances.

For example, the way you squeeze determines which of the fluids are recorded. If a stack of papers is used, the more remote sheets will be affected only by the portions of the leaf that have plentiful quantities of nonviscous fluids such as sap held in cell cavities.

Chemical separations can be used in at least two ways. One is to use selective washings after the filter-paper impressions are made but before the photographic exposure. For example, washing with trichloroacetic acid leaves behind the proteins and nucleic acids. Another separation technique is to prepare the papers with precipitators in advance.

The method was described at the Paris UNESCO conference (Abstract No. 150) on radioisotopes by its inventor, Z. A. Medvedev of Russia. He has applied it to localization of protein and nucleic-acid synthesis in bean plants.

Can Gammas Kill Germs for Hospitals?

Gamma photons have recently been proposed as a cheaper means of hospital sterilization. If they can be proved to be money savers, they would somewhat relieve the constant pressure of hospital expenses. Take blankets, for example. They must now be sterilized in a steam autoclave and this sterilization wears them out. The thorough treatment required for surgical bedding destroys it in two or three applications.

At Harwell, England, Dr. Henry Seligman is working on radiation sterilization—a method that might extend bedding life by a factor of 10. Needless to say, the method, once developed, might have economic applications outside hospital wards.

Instrument sterilization. In another direction John P. Herrlin and Vito Pagano, M.D.'s, of New York City's Bird S. Coler Hospital have been exploring the use of radiation for sterilization of surgical instruments. Accelerators, they find, are inappropriate for the problem because of limited penetration of charged particles. The cost is too high, and the dose that would sterilize would cause deterioration of material. A better possibility appears to be a gamma chamber in which a relatively low level of radiation could keep sterile such instruments as are used for a short time, stored for a long time. In food studies it has been demonstrated that a surprisingly low level of gamma radiation will inhibit bacterial growth when the exposure is constant. (NU, Nov. '57, p. 175). This type of treatment, on the other hand, will not meet one important surgeon's need: quick sterilization of a one-of-a-kind instrument that falls on the floor during an operation

In preliminary investigations of this subject Herrlin and Pagano have found that a surprisingly small amount of attention has been given to it. It is their plan to remedy the situation with further investigations.

Carrier Content by Neutron Activation

Neutron-activation analysis has been used for confirmation of the carrier-free status of radioisotopes and determination of specific activity of isotopes that are not carrier free. Richard P. Spencer, Thomas G. Mitchell, and E. Richard King, have developed the method at the U. S. Naval Hospital, Bethesda, Md.

The technique is to measure the radioactivity of a sample, expose it to thermal neutrons, and then measure the activity again. Increase in the activity can be used to identify and measure carriers.

For example, how much carrier Cu^{63} is present in a sample of Cu^{64}? Disintegration of Cu^{64} produces Zn^{64} and Ni^{64}, and n,γ reactions in the four nuclides known to be present produce Cu^{64}, Cu^{65}, Zn^{65}, and Ni^{65}. Of these the only nuclide produced in detectable amounts is Cu^{64}. Hence the test requires only an evaluation of the increase in Cu^{64} activity.

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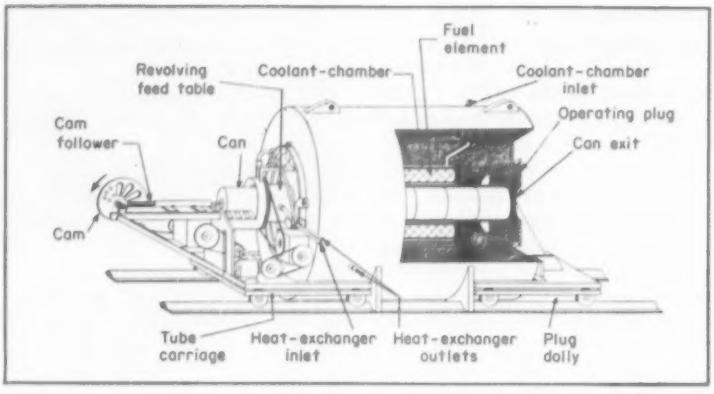

FIG. 1. HIGH-LEVEL IRRADIATION FACILITY has automatic feed mechanism that pushes cans through cask holding 8 MTR fuel elements. Dose is adjusted by speed of feed mechanism. Can that takes 1 hr in transit receives 3 X 10^6 r

Dugway Food Irradiator Uses 8 MTR Elements

By CARL J. CHRISTENSEN, ANDREW JACOBSEN, QUENTIN KLINGER and GEORGE ABEL
University of Utah, Salt Lake City, Utah

Eight spent MTR fuel elements are the radiation source in the Dugway high-level gamma irradiation facility, designed especially for canned-food processing (Fig. 1). It was designed and built by the University of Utah for the Food and Container Institute of the Army Quartermaster Corps.

The system will accept any can up to the No. 10 size and deliver a specified, uniform radiation dose between 25,000 and 10^7 r. Variation in total dose is limited to a few per cent. Provision is made for temperature control in the range of 25–200° F.

Conveyor

When the facility is in use, cans of food are automatically conveyed to a position in front of a feed tube running through the cask that holds the spent fuel elements. A cam-and-follower system shoves the cans through the feed tube, which rotates to improve dose uniformity. Part of the conveyor system can be seen in Figs. 1 and 2.

A chain belt lifts the cans from their storage cabinet to the top of a chute. As each can drops into the chute it turns through 90 deg and rolls into place in front of the cam follower. The follower is a spring-loaded plunger that advances in response to the cam rotation and pushes the can into the feed tube. Then the next can rolls into place, and its insertion advances the can before it as well as the other cans in the tube. After a can has completed its trip through the radiation field, it rolls down a chute and back into the storage cabinet.

The conveyor system is driven by a variable-speed motor. A small reversible motor adjusts the speed, which controls the radiation dose by changing the time that each can spends in the radiation field. The fuel elements currently in use deliver $\sim 3 \times 10^6$ r to a No. 10 can that takes 1 hr to make its transit through the irradiation chamber.

The facility is now equipped to handle No. 2 and No. 10 cans. When the smaller cans are to be processed, one inserts a tube of suitable diameter inside the larger feed tube and makes minor adjustments in the conveyor.

Shielding Cask

The cask that holds the fuel elements serves as both a radiation shield during use and a shipping container. It is a steel-enclosed lead cylinder with a fuel-element basket inside. Concrete cast into the corners of the cylinder before the lead was poured lightens the cask and reduces its cost. Total weight is 12½ tons. The cask is shown in Figs. 3, 4, and 5.

The stainless-steel fuel-element basket is surrounded by a coolant chamber in which a coolant can be circulated. Its purposes are to cool the interior during shipment and partially control the temperature of the irradiation chamber during use.

A loading plug is removed only at the reactor when fuel elements are to be exchanged. In addition there are interchangeable shipping plugs and operating plugs at each end of the cask. They are stepped and tapered to prevent gamma-ray streaming. Through the operating plugs are holes to permit the insertion of the feed tube. Neoprene O-rings make the bottom shipping plug watertight.

Two remotely operated dollies like the one of Fig. 6 are used to insert and remove shipping and operating plugs. When a shipping plug is in place, the shielding is good enough that a workman can bolt the plug to the cask or the dolly. When plugs are to be exchanged, the dolly slides away from the cask on tracks and is removed from

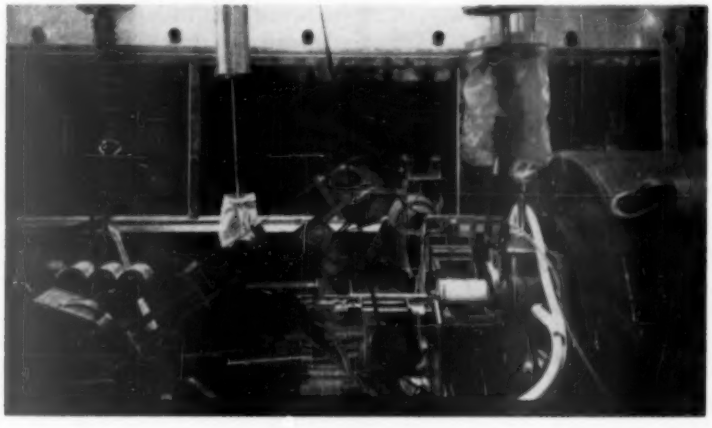

FIG. 2. CHAIN-BELT CONVEYOR lifts cans to top of chute, which guides them into place in front of cam follower. Four cans at far left are water-filled dummies that precede and follow cans processed in a run

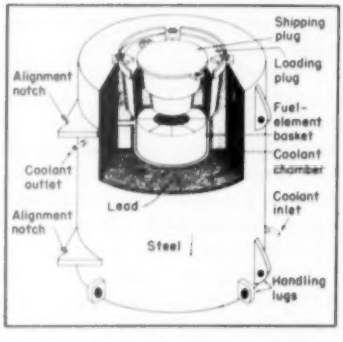

FIG. 3. SHIELDING AND SHIPPING CASK is lead-filled steel shell. Stainless-steel basket holds fuel elements during shipment and operation

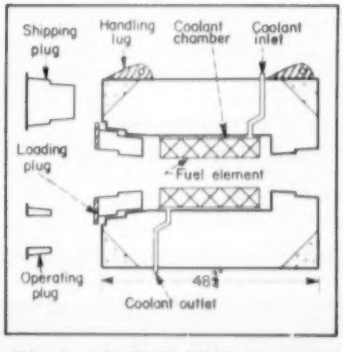

FIG. 4. LOADING PLUG is removed only at reactor for exchange of fuel elements. Shipping plugs are interchangeable with operating plugs

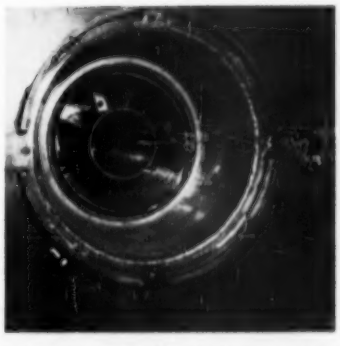

FIG. 5. INTERCHANGEABILITY of shipping plugs and operating plugs with loading plug bolted in place is shown in this end view of cask

FIG. 6. DOLLIES for PLUG HANDLING can be remotely moved on tracks that hold cask and removed from cave by remotely operated crane

FIG. 7. COOLING COILS are visible as handling mechanism is withdrawn from cask. In combination with coolant chamber they enable temperature control over 25°–200° F. Intense radiation field requires auxiliary heat exchanger between coils and compressor. Dehydrated ethyl alcohol is refrigerant

the facility by an overhead crane. The new plug is substituted in the dolly for the one that has been withdrawn, and the remotely controlled operation is reversed to insert the new plug.

Shielding. The operating plugs limit the escape of gamma rays from the ends of the facility during operation. When the feed tube is full of cans, radiation outside the cask is limited to 1–2 r/hr except for two small cones about the axis. Here the intensity is sometimes as high as 300 r/hr. Thus in an emergency a person could work for a short time in the area around

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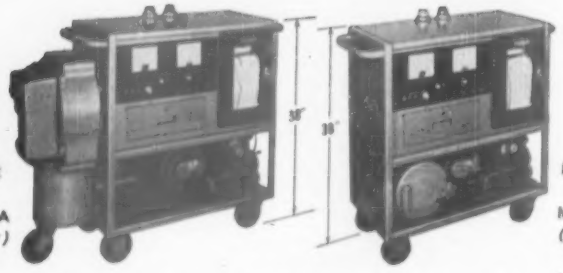

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the cask if he avoided the cones at the end.

With shipping and loading plugs in place and eight 70-day fuel elements in the cask, no radiation above background can be detected at the side of the cask. A few milliroentgens per hour have been detected at the ends. This is due either to leakage of photons through the channels between cask and plugs or to a small amount of slightly radioactive canal water that remains in the cracks after the cask has been lifted from the canal and washed.

Temperature Control

The facility has two systems for temperature control. One is the coolant chamber that is part of the cask. The other is a cooling coil that occupies a position between the line of cans and the fuel-element basket. It is shown in Fig. 7.

The coolant chamber is available for cooling during shipment and can be connected to an external heat exchanger. However, computation and experience show that with 70-day elements no external exchanger is necessary during shipment. If the cask is filled with water, there is enough heat conduction to the walls to keep the fuel-element temperature within 100° F of the ambient. Thus the water never boils.

This coolant chamber is not adequate for the 25–200° F temperature control specified for food processing. Therefore the coils of Fig. 7 were soldered to a cylindrical shell. The inside diameter is such that the feed tube can be inserted, and the outside diameter allows a snug fit inside the fuel-element basket. This heat exchanger is permanently fastened to the dolly that holds the feed tube and has inlet and outlet tubes for circulation of fluid.

After some seriously disappointing experiences with radiation damage to heat-exchanger fluids and subsequent damage to the compressor, we inserted a second heat exchanger between the compressor and the temperature-control exchanger. Dehydrated ethyl alcohol was then introduced as the refrigerant and has been satisfactory, although not free of difficulties. The radiation causes some chemical changes, and aldehydes, esters, acids, and other chemical species are formed in moderate quantities. Work is now under way to distinguish these degradation products. However, ethyl alcohol can be used for several months before being cleaned up.

BOOKS

Molecular Beams

By NORMAN F. RAMSEY (Oxford University Press, New York, 1956, xii + 466 pages, $12.00).

Reviewed by I. I. RABI, Department of Physics, Columbia University, New York, N. Y.

Ramsey's "Molecular Beams" is the first comprehensive book to appear on this subject since Fraser's monograph "Molecular Rays," which was published in 1931. This is surprising since the field has had a large expansion in the intervening quarter century. The explanation may lie in the fact that this demanding discipline left the prospective authors no time to describe the field as a whole.

In 1931 only one major laboratory, Otto Stern's in Hamburg, the fountainhead of molecular-beam lore, was in existence. Now at least ten active groups, chiefly in the United States but also in Germany and England, are engaged in research using the classical molecular-beam techniques in which the atoms and molecules are detected directly. The main effort has gone into the study of nuclear properties such as spin, magnetic moment, electrical quadrupole moments, magnetic octupole moments of stable and radioactive nuclei. Most of what is known of these important nuclear properties derives from molecular-beam experiments.

Some of the most basic discoveries in twentieth-century physics have been made with molecular-beam techniques. Since 1931 there have come the anomalous magnetic moments of the proton and neutron, the electric quadrupole moment of the deuteron, the intrinsic magnetic moment of the electron and the shift of the *S*-state energy levels in hydrogen. Only the fields of cosmic rays and high-energy physics with vastly larger number of workers can match this record of the last 25 years.

The successful application of radio-frequency resonance techniques has, in the last 25 years, radically changed the objectives and scope of molecular-beam experiments. Results that were formerly obtained with great difficulty and with poor or only moderate precision can now be obtained with ease and with precision that is hardly exceeded in any other field of experimental physics. A recent example of the application in precision methods is the cesium clock, which promises to have precision and accuracy orders of magnitude beyond what has been previously available. Offshoots of the molecular-beam method, such as the maser and microwave spectroscopy as well as nuclear induction methods, are now fields of research of greater activity than the classical molecular-beams field itself.

One important field of research that originated with Stern in Hamburg has unfortunately languished. This is the use of molecular beams in the sense of matter waves and the physical optics connected with these phenomena, both in themselves and as a tool for the study of other properties of matter such as surfaces. Modern methods of detection should make this field more accessible and fruitful.

Ramsey's book covers the whole field with great detail and clarity. Student and expert alike will find it indispensable. All the necessary theory is contained within the cover of this volume. A knowledge of the elements of quantum mechanics is of course presupposed in the exposition of the theory behind molecular-beam experiments, but the requirement is not greater than that given in the usual introductory course. The book is thoroughly up-to-date and the excellent bibliography offers easy access to the literature of the subject.

Quite naturally the author stresses the work to which he and his students at Harvard have devoted themselves, but this only gives added interest and a personal touch to this timely and highly successful presentation.

Magnetohydrodynamics

Edited by ROLF K. M. LANDSHOFF (Stanford University Press, Stanford, 1957, x + 115 pages, $4.00).

Reviewed by H. L. DAVIS, NUCLEONICS.

Anyone who can define magnetohydrodynamics (the study of dynamics of high-temperature ionized gases in the presence of magnetic fields) is aware that controlled fusion is the ulterior motive behind the sudden widespread interest in this formidable topic. Those, however, who read this book expecting advice on how to build a controlled-fusion device will be disappointed. The closest the book comes to answering this hope is a theoretical discussion of the pinch effect; otherwise it is pretty much dedicated to understanding the basic physics of a high-temperature plasma interacting with a magnetic field.

Although the level is appropriate for serious students, a casual reader will pick up some of the flavor of this brand new subject provided he has a strong background in electromagnetic theory plus a little knowledge of hydrodynamics.

The book is based on the contributions presented at a Magnetohydrodynamics Symposium sponsored by Lockheed Missile Systems Division at Palo Alto in December, 1956. As such, the book consists essentially of twelve papers by various authors in the field. The editor has attempted to rework the material into an integrated form and has chosen papers that are representative of the different areas of interest in the field. The result does not serve as a textbook development of the subject but rather as a short sampler.

The volume is divided into two well-defined sections. The first treats the theory of magnetohydrodynamics. The introductory article by A. R. Kantrowitz and H. E. Petschek uses order-of-magnitude arguments involving characteristics lengths to arrive at an over-all classification of the various regions of interest. The four remaining articles are concerned with specialized topics in the theory. The experimental section is introduced with a short discussion by the editor about dimensionless parameters in magnetohydrodynamics. Following this there are six papers describing experiments in the field.

Introduction au Génie Nucléaire

Vol. 1: Physique et Calcul des Réacteurs Nucléaires
Vol. 2: Contrôle et Protection des Réactéurs Nucleaires

By THÉO KAHAN and MAURICE GAUZIT (Dunod, Paris, 1957; Vol. 1—xiv + 388 pages, 3900 francs; Vol. 2—xiv + 396 pages, 3900 francs).

Reviewed by THEODORE BERESOVSKI, Nuclear Products—ERCO Division of ACF Industries, Inc., Washington, D. C.

The authors have compiled in the first two volumes of this introduction to nuclear engineering a great deal of useful information on reactor theory and technology. They have succeeded in rewriting and organizing this international material into a clear, well developed book on reactor-engineering fundamentals.

The first volume can be divided into

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two sections, each containing three chapters. The first section deals with atomic and nuclear physics and the second with reactor theory and calculations. The second volume consists of four chapters on the subjects of radiation detection, reactor control, dangers of ionizing radiation and protective shielding. It has also a 24-page appendix on fusion reactions. The third volume, presumably on power reactor technology, has been announced but has not yet been published.

This is the first book of this type that has been published in France, and undoubtedly it will be of great interest to all nuclear-minded Frenchmen. Its value in the United States will be somewhat lessened by the language barrier and by the existence of several earlier texts, such as S. Glasstone's and M. C. Edlund's "The Elements of Nuclear Reactor Theory," R. Stephenson's "Introduction to Nuclear Engineering," S. Glasstone's "Principles of Nuclear Reactor Engineering," etc., which also present similar material.

The text thoroughly covers the topics that are treated, and it compares quite favorably with the above-mentioned books. Many mathematical derivations are presented in detail and a number of problems are worked out numerically. Also, numerous graphs and tables present a great deal of useful design information. For those who would like to pursue the various subjects further, extensive reference lists are provided.

Occasionally one might wish that the authors had included discussions of additional items. In vol. 1, for instance, when they numerically describe a gas-cooled homogeneous reactor, it would appear that they should have either developed the problem further or postponed its discussion until vol. 3. Similarly, consideration of reactor disaster clouds and their atmospheric diffusion would seem to be called for in vol. 2, which is devoted exclusively to the subject of reactor control and protection.

By and large, however, these are excellent introductory books and a welcome source of reference information for nuclear engineers. The third volume will be awaited with great interest.

BOOKS RECEIVED

The Industrial Challenge of Nuclear Energy (The European Productivity Agency of the Organization for European Economic Co-operation, Paris, 1957, 288 p., $5.00). This book presents the proceedings of the first Information Conference on Nuclear Energy for Management held in Paris, April, 1957. The international panel of speakers presented papers on technical, economic and industrial aspects of nuclear energy, industrial uses of radioisotopes, and the development programs being pursued in various countries.

Effect of Radiation on Human Heredity, World Health Organization (Columbia University Press, New York, 1957, 168 p., $4.00). A study group organized by the World Health Organization met in Copenhagen, August, 1956, to consider genetic effects of ionizing radiations on man. This book contains the report of the study group as well as papers presented by various members. Among the group's recommendations are additional institutions devoted to the study of genetics, more training and public education in genetics, registration of serious hereditary diseases and defects, investigations to study radiation damage to tissues and on effects on life span, and, as an approach to control of radiation, records of individual exposures.

Transistor Electronics, by A. Lo, R. Endres, J. Zawels, F. Waldhauer, and C. Cheng (Prentice-Hall, Inc., Englewood Cliffs, N. J., 1957, 520 p., $12). This book provides a basic understanding of transistor operation. Discussion covers high-frequency operation and practical circuitry.

ALSO OF NOTE

AEC health and safety studies. The following reports are available: KAPL Air Cleaning Program (KAPL-1014, Jan. '57, 17 p., 20¢), Semiannual progress Report of Radiological Development Activities in the Health Physics Unit (KAPL-1572, July–Dec. '55, 36 p., 45¢), Measurement of the Scatter Component from a Kilocurie Cobalt-60 Source (NYO-2065, June '57, 13 p., 20¢), The Metabolism of Indium-114m Administered to the Rat by Intratracheal Intubation (UR-500, July '57, 46 p., 35¢). *Office of Technical Services, U. S. Department of Commerce, Washington 25, D. C.*

U. S. Research Reactors. More than 30 research reactors in operation or

under construction are described in this booklet prepared by Battelle Memorial Institute. One or two examples are given for major reactor types. Data includes cost information. 73 p. *Office of Technical Services, U. S. Department of Commerce, Washington 25, D. C., $1.50.*

Geneva Conference on microcards. A microcard edition of the entire proceedings of the first International Conference on the Peaceful Uses of Atomic Energy held in Geneva, 1955, is available. *The Chronicle of United Nations Activities, 234 W. 26 St., New York 1, N. Y., $70.*

Technical reports available. An Analysis of Temperature Coefficients of Solubility in Dilute Liquid Metal Solutions, by S. Strauss, J. White, B. Brown, Naval Research Laboratory (PB-131065, 23 p., 75¢); Measurement of Radon Gas in Cylindrical Ionization Chambers, by E. Burke, Wright Air Development Center (PB131186, 11 p., 50¢); Theoretical Analysis of the Response of a Proton Recoil Type Neutron Detector, by S. Hartmann, Wright Air Development Center (PB131204, 55 p., $1.50). *Office of Technical Services, U. S. Department of Commerce, Washington 25, D. C.*

The Chemical Analysis of Zirconium and Zircaloy Metals (NMI-1171). The methods presented here for the analysis of impurities in zirconium and its alloys are the result of a program in which eleven laboratories participated. Chemical and spectrographic methods are described. 108 p. *Office of Technical Services, Department of Commerce, Washington 25, D. C., 60¢.*

Atomic Energy as a Means of Economic Development—A Selected Bibliography. A cross section of general, semitechnical and technical bibliographies, books and articles are listed. Official (UN, governmental, etc.) and unofficial sources included. 40 p. *C. G. Alexandrides, 2 Tudor City Place, New York 17, N. Y., $1.00.*

Career Satisfactions of Professional Engineers in Industry. Based on interviews with professional engineers, this survey indicates what they like and dislike about their career environments. The survey was made by Opinion Research Corp. 84 p. *Professional Engineers Conference Board for Industry, 2029 K St., Washington 6, D. C., $3 for nonmembers.*

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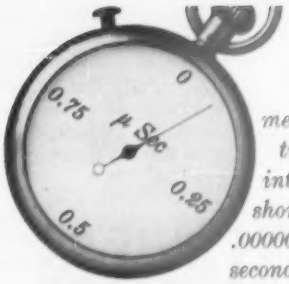

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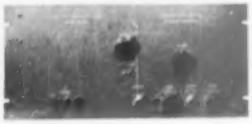

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NUCLEAR CALENDAR

Jan. 27-28—Sixth Scintillation Counter Symposium, AEC, AIEE, IRE and NBS, Washington, D. C. (*Shoreham*). Contact G. A. Morton, RCA Laboratories, Princeton, N. J.

March 2-6—American Society of Mechanical Engineer's gas turbine power division conference and exhibit, Washington, D. C. (*Shoreham*). Contact L. S. Dennegar, ASME, 29 West 39 St., New York 18.

March 17-21—Nuclear Congress will include four events, at International Amphitheater, Chicago: **A.** Fourth Nuclear Engineering & Science Congress (EJC), running all five days. Contact congress manager, American Institute of Chemical Engineers, 25 West 45 St., New York 18. **B.** Atomic Industrial Forum Atom Fair, all five days, contact J. V. Friel, 304 Architects Bldg, Philadelphia 4. **C.** Sixth Hot Laboratories & Equipment Conference, 19-20th; contact Frank Ring, Jr., ORNL, Oak Ridge, Tenn. **D.** Sixth Atomic Energy in Industry Conference, National Industrial Conference Board, 17-19th; contact John J. McMahon, 460 Park Ave., NYC.

April 20-23—Canadian—U. S. chemical engineering conference, American Institute of Chemical Engineers and Chemical Institute of Canada, Montreal. Contact H. R. L. Streight, DuPont Co. of Canada, PO Box 660, Montreal, Quebec, Canada.

May 27-29—Third annual reactive metals conference, American Institute of Mining, Metallurgical and Petroleum Engineers, Buffalo (*Statler*). Contact S. F. Urban, National Lead Co., Buffalo, N.Y.

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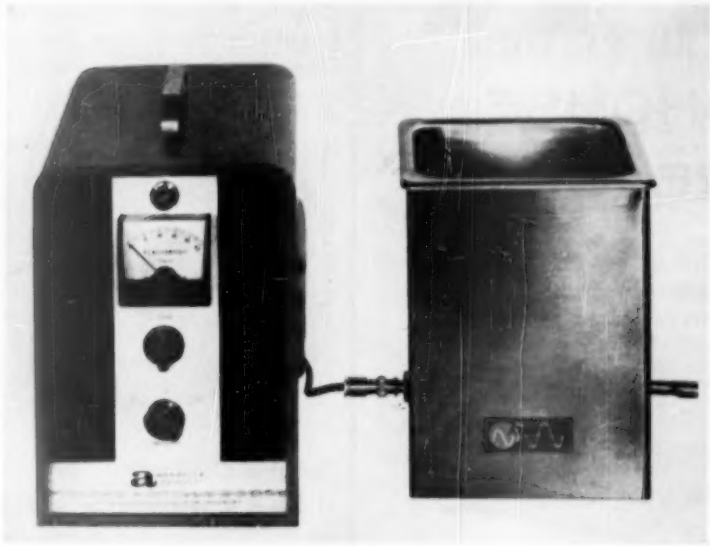

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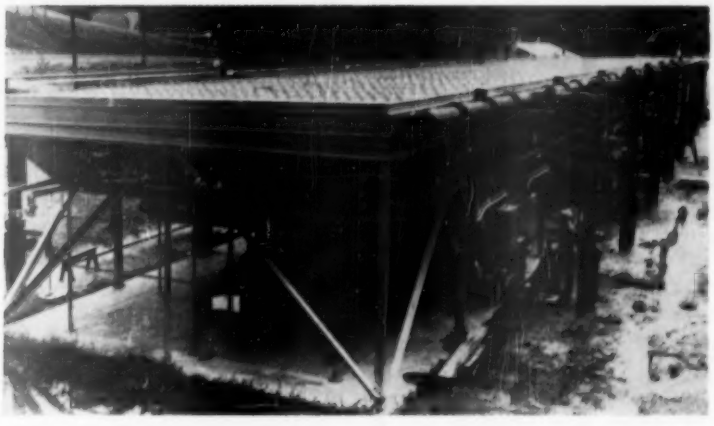

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Isolite, a lamp illuminated by radioactive Kr^{85}, provides power-free, uninterrupted light up to 10 years. Light source is composed of phosphor crystals, enclosed in a hermetically-sealed transparent capsule and excited to luminescence by the gas.—U. S. Radium Corp., Morristown, N. J.

Harmonic Drive Control Rod

Harmonic drive principle of motion transmission permits hermetically sealed means of controlling reactors (above). Rotating parts are exterior to pressure vessel while internally there is one moving part. Positive position indication is accomplished by a mechanical indicator that is directly connected and responsive to control-rod position. Driven by conventional power source, it has these characteristics: normal operating pressure, 2,200 psi; normal operating temperature, 565° F, with max operating tempera-

Correction. Channel capacities for PA-100 pulse-height analyzer are 10^5 counts per channel-rather than 10 as stated previously (NU, Nov. '57, p. 192).—Technical Measurement Corp., 140 State St., New Haven 11, Conn.

WTR

WESTINGHOUSE TESTING REACTOR

Will be available to industry for in-pile experiments approximately April 1, 1959. This reactor will provide a new complete irradiation service featuring thermal neutron fluxes up to 10^{14}/n/cm²-sec at temperatures and pressures as high as 600°F and 2000 psi. Auxiliary services including rabbit facilities, hot cells, and gamma irradiations are also available.

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P. O. Box 1075, Pittsburgh 30, Pa.

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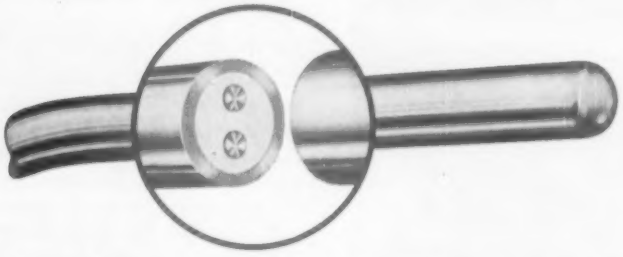


Designed for use under extremely severe conditions, enclosed hot-junction "Ceramo" thermocouples—with ceramic insulated, metal sheathed wire—will outlast comparable standard types many times. And response time is just as fast. Furthermore, both sheathing and insulation can be specially selected for resistance to atomic radiation. Conductors in all standard calibrations and various sheath materials are available for temperatures up to 3000° F. If desired, "Ceramo" thermocouples can be completely pre-tested. Conductors—36 to 12 gage. Overall diameters—1/25" to 7/16".

Write for Bulletin 305-K.

Thermo Electric Co. Inc.
SADDLE BROOK, NEW JERSEY
In Canada—THERMO ELECTRIC (Canada) Ltd., Brampton, Ont.

ture, 650° F; linear speed of control rod, 20 in./min; length of travel, 30 in.; max permissible load at 70° F, 74,400 lb and max permissible load at 650° F, 67,200 lb.; .5 hp motor.—Atomic Power Department, United Shoe Machinery Corp., Beverly, Mass.

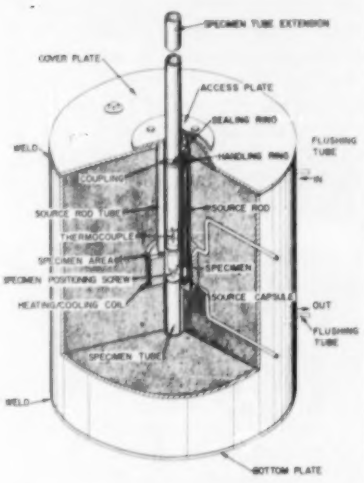
Portable Oscillator

Model 20A portable test oscillator (above) is continuously variable over a frequency range of 15 cycles to 150 kc at a source impedance of <0.5 ohm with output constant to ±2%. Oscillator has all-transistor circuitry, printed wiring and self-contained power supply.—Alectra Division, Consolidated Electrodynamics Corp., 325 N. Altadena Drive, Pasadena, Calif.

Self-Contained Irradiator

Model A, the Hotrodder (above), can hold up to 25,000-c Co^{60} and is a complete installation in itself. Unit operates this way: sample is inserted into irradiation chamber; operator pushes single button on control console; sample door closes; source moves into position automatically. At end of preset time cycle reverses. Source mechanism and door leading to irradiation chamber are fitted with interlocks that prevent source being moved to exposed position while sample door is open. Irradiation chamber will hold sample up to 10 in. in dia by 10 in. in length.

Max surface dose rate is 50 mr/hr.—Nuclear Systems, A division of the Budd Co., Philadelphia 32, Pa.



Gamma Source Container

Designed for use as a standard item of laboratory equipment, Gammalab container (above) is equipped with 62-curie Co^{60} source with an average dose rate of 100,000 r/h. Gammalab is less than 2 ft high and takes a 3.2-ml sample volume.—Radiation Applications, Inc., 342 Madison Ave., New York 17.

Decade Scaler

Model DS-1A (above), decade scaler for precision counting and registering of electrical impulses, is equipped to power all basic nuclear detectors and count their output at rates to 10^6 cpm. Counting periods can be controlled by built-in preset timer to accuracy of 0.2 sec in 55 min. Has test signal of 3600 cpm.—Nuclear Measurements Corp., 2460 N. Arlington Ave., Indianapolis 18, Ind.

Rod-Position Indicator

Control-rod position indicator provides an output voltage of 2–6 volts/ft that

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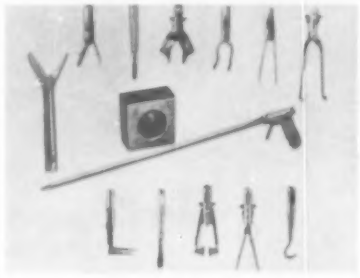
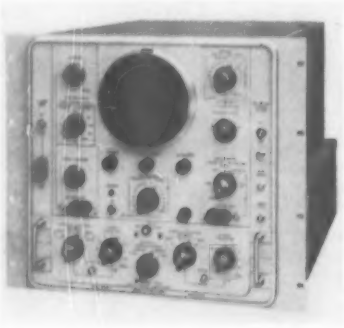
is proportional to position and is linear within 0.1% over a 10-ft stroke. Unit consists of three mechanically separate assemblies, a magnetic core and tubular coil assembly separated by a pressure wall. Magnetic probe is attached by stainless-steel tube to control rod. Separate coils at 0- and 10-ft positions actuate limit relays. Pressure wall will withstand 1,200-psi steam at 650° F.—G. L. Collins Corp., 2820 E. Hullett St., Long Beach 5, Calif.



Remote Handling Tongs

Ball-Swivel Radiarm (above), a remote handling tong, is designed for working with high-level radiation within shielded caves, hot cells and irradiation chambers. Tong is equipped with slots to permit attaching different heads for lifting, picking, clamping, etc. The swivel is lead-shielded for mounting directly into the protective wall.—Atomic Center for Instruments & Equipment, Inc., 489 Fifth Ave., New York 17, N. Y.



General-Purpose Oscilloscope

Model 150AR d-c to 10 mc oscilloscope is mounted on slides for accessibility and features calibrated sweep magnifications of 5, 10, 50, and 100; any portion of the magnified trace may be viewed. Unit is used with plug-in vertical amplifiers.—Hewlett-Packard Co., 275 Page Mill Rd., Palo Alto, Calif.

Scintillation Transducer

New unit (above) combines a NaI(Tl) crystal package with a standard 5-in. photomultiplier tube. Crystals are produced in new facility with emphasis on uniformity of Tl distribution for uniform response to γ rays stopped. Available in crystal lengths up to 4 in., the packaged detection units are housed in magnetic shields and come complete with preamplifiers.—Levinthal Electronic Products, Inc., Stanford Industrial Park, Palo Alto, Calif.

Warning Paint

A daylight fluorescent paint has been developed for increasing effectiveness of the radiation warning symbol. Called Radiation Purple Sunbonded Day-Glo, this AEC-approved paint is brighter in daylight than nonfluorescent purple, besides being distinguishable at a distance. Available in aerosol spray cans, as a brush-or-spray formulation and for silk-screen work.—Switzer Bros., Inc., 4732 St. Clair Ave., Cleveland 3, Ohio.

New Van de Graaff

Model GS 1.5-Mev Van de Graaff electron accelerator was specially designed for radiation processing. It is rated at 2.5 kw with peak-power range of 4-5 kw. Penetration through $\frac{5}{16}$-in. is obtained at a unit density. Unit is 96 in. high and extends 66 in. below the base plate. Electron-beam width controllable from 5 to 15 in.—High Voltage Engineering Corp., Burlington, Mass.

LITERATURE AVAILABLE

Flow-pilot valves in five basic models—ball cam, palm button, hand lever, locking hand lever and mechanical link clevis—are described in catalog 262. 8 p.—Hanna Engineering

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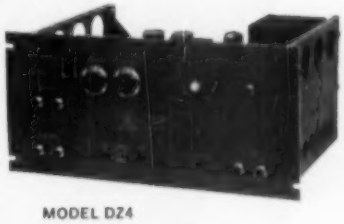
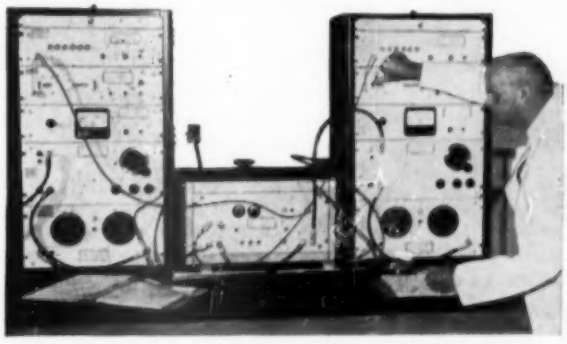


MODEL DZ4



DZ4 in use with two DZ41 Gamma Ray Spectrometers. Physics Department, Ohio State University (Courtesy, Ohio State University).

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Conversion of its plant to SX is an excellent demonstration of Vitro's know-how across the spectrum of modern technology. Engineers from Vitro Engineering Company, scientists from Vitro Laboratories, and construction specialists from The Refinery Engineering Company joined with Vitro Uranium Company and Vitro Rare Metals Company to build this better mousetrap, uranium style.

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Works, 1765 Elston Ave., Chicago 22, Ill.

Automatic indexing X-ray spectrograph, called the Autrometer, is described. 8 p.—Instrument Division, Philips Electronics, Inc., 750 S. Fulton Ave., Mt. Vernon, New York.

Gadolinium oxide and nitrate and their properties are described in data sheet 301.—Research Chemicals, Inc., PO Box 431, 170 W. Providencia St., Burbank, Calif.

Radiochemical techniques used in obtaining technical data, are described with case histories in bulletin 408.—Nuclear Science & Engineering Corp., PO Box 10901, Pittsburgh 36, Pa.

Plastic fluor for scintillation counting is described and typical applications given. 4 p.—Cadillac Plastic & Chemical Co., 15111 Second Ave., Detroit 3, Mich.

Medical isotope procedures. A series of papers and reports covering areas in which greatest isotope progress has been made.—Abbott Laboratories, North Chicago, Ill.

High-vacuum gages plus high-vacuum valves, baffles and traps are listed in two bulletins—9-1 and 10-1.—Rochester Division, Consolidated Electrodynamics, Rochester 3, New York.

"Nuclear Safety" discusses firm's scientific and engineering service in reactor and radiation hazard evaluation. 12 p.—Holmes & Narver, Inc., 828 S. Figueroa St., Los Angeles, Calif.

"Instrumentation and Control of Nuclear Plants" (booklet B-7233) covers these subjects—power control, instrumentation, coolant instrumentation and radiation monitoring. 8 p.—Westinghouse Electric Corp., PO Box 2099, Pittsburgh 30, Pa.

Commercial-grade zirconium with its applications and properties is discussed in booklet that reviews established and potential applications, gives corrosion and physical properties and discusses methods of fabrication.—Columbia-National Corp., 70 Memorial Drive, Cambridge 42, Mass.

Beryllium oxide is subject of folder that includes discussions on powder and crucibles, crucible wash and shapes.

8 p.—The Beryllium Corp., PO Box 1462, Reading, Pa.

S^{35} and P^{32} labelled compounds are listed in catalog. 4 p.—Volk Radio-Chemical Co., 5412 N. Clark St., Chicago 40, Ill.

Russian journals available in English translation listed in catalog #2. 64 p. —Consultants Bureau, Inc., 227 West 17 St., New York 11, N. Y.

Components for liquid-metal systems described in six new bulletins: sodium-oxide control station in bulletin GEA-6453A; liquid-level indicators in GEA-6454A; cold traps in GEA-6456A; plugging indicator in GEA-6457A; magnetic flowmeter in GEA-6360A; and a-c electromagnetic pump in GEA-6359A.—General Electric Apparatus Sales Division, Schenectady 5, N. Y.

INDUSTRY NOTES

▶**National Beryllia Corp.** has opened a new production plant in Haskell, N. J. to fabricate precision ceramics.

▶**Radiation Counter Laboratories, Inc.,** Skokie, Ill. is doubling its plant space with a 6400 ft² addition.

▶**Radiation Instrument Development Laboratories,** Chicago, has tripled its production space and facilities.

▶**Tracerlab, Inc.** now has trained decontamination squads operating from its three base locations—Waltham, Mass.; Richmond, Calif. and Houston, Tex.

▶**M&C Nuclear, Inc.** has been formed as a wholly-owned subsidiary of Metals & Controls Corp. The former nuclear products division's present management will become officers of the new subsidiary.

▶**Flanders Mill, Inc.,** Riverhead, N. Y., manufacturer of filters for nuclear applications, has changed its name to Flanders Filters, Inc.

▶**Applied Research, Inc.,** Chicago, an industrial consulting firm engaged in nuclear research, has changed its name to Booz, Allen Applied Research, Inc., with new offices at 135 S. LaSalle St., Chicago.

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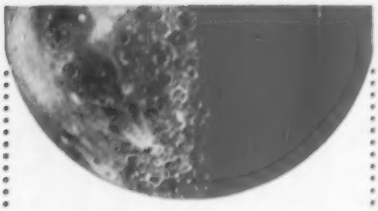

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5. *Senior Engineer Research or Engineering Specialist:* PhD. desired with a background in nuclear energy, servo mechanism, controls or instrumentation. Should have preferably 4 years' additional experience.

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7. *Senior Structural Engineer or Stress Analyst:* Several years' experience in mechanical stress and structural analysis. M.S. or B.S. degree or equivalent.

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NEWSMAKERS

Westinghouse awarded two metallurgists—**Donald E. Thomas** and **Kenneth M. Goldman**—special $2,000 prizes for their discovery of a broad range of zirconium alloys which they developed originally for the structural parts of the *Nautilus* reactor and the cladding of certain components.

Peter Fortescue, who served with Britain's Atomic Energy Research Establishment for 10 years, has joined General Dynamics' General Atomic division. At GA he will concentrate on experimental applications of new engineering concepts. While with AERE at Harwell Fortescue helped set up a laboratory to handle engineering problems associated with power reactor and diffusion plant programs, including development of cooling arrangements at Calder Hall.

S. Allan Lough has been appointed director of AEC's health and safety laboratory in the New York Operations Office, succeeding **Merril Eisenbud,** who continues as manager of NYOO.

Union Carbide Nuclear Co. has promoted **Sylvan J. Cromer** to vice-president for engineering. Cromer, who joined Carbide in 1944, has held many engineering posts at Oak Ridge, the most recent as chief engineer of the gaseous diffusion plant's engineering division.

C. D. W. Thornton, who joined Farnsworth Electronics Co. in 1956 as assistant to the president for atomic energy, has been appointed director of research and development.

Argonne National Laboratory has appointed **Paul K. Kuroda** to its radiological physics staff. Kuroda, a native of Japan, had been associate professor of chemistry at the Univ. of Arkansas since 1955.

Kenneth A. Dunbar has been appointed manager of AEC's Chicago Operations Office, succeeding **John J. Flaherty,** who left to join Atomics International. Dunbar, with the atomic energy program since 1943, managed both the Paducah and Portsmouth Operations Offices while those facilities were being built. He received AEC's Outstanding Service Award in 1956.

Stone & Webster Engineering Corp. has named **Charles R. Claxton** assistant chief engineer of its nuclear projects division.

Beryllium Corp. has appointed **Edmund M. Velten** technical director. A specialist in nuclear metallurgy, he has been with the atomic energy program since the Manhattan District.

The Atomic Industrial Forum elected six new directors, to serve until 1960: **James B. Austin,** vice president for research and technology, U. S. Steel Corp.; **Marion W. Boyer,** director, Standard Oil Co. (New Jersey) and former AEC general manager; **J. Victor Herd,** chairman and president, American Fore Insurance Group; **Henry D. Smyth,** board chairman of scientific and engineering research at Princeton Univ. and former AEC commissioner; **T. Cortlandt Williams,** president, Stone & Webster Engineering Corp.; and **Hood Worthington,** E. I. du Pont de Nemours & Co.

Wendell A. Horning has joined Atomics International as group leader in theoretical neutron physics. Horning previously had been with Ramo-Wooldridge Corp., where he worked on the analysis of reactor test data, and with General Electric at Hanford.

F. E. Jablonski has joined General Motors' nuclear power engineering department. He formerly headed the neutron physics section at the Naval Research Laboratory, Washington, D. C.

Nuclear Materials & Equipment Corp. has appointed **Francis M. Cain, Jr.** manager of metallurgical services.

David Bandel has been named associate technical director of Tracerlab, Inc. Bandel, a chemist, had been with AMF for the past six years. He succeeds **William Tittle,** who has become a professor at Southern Methodist University where he is developing a reactor engineering program.

North Carolina State College has named **Harold A. Lamonds** reactor supervisor, and **E. Jack Story** health physicist for its reactor facility staff.

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• The Commission has also established a program of financial assistance to colleges and universities for studying radiobiology and use of radioisotopes in life science courses. Grants, not to exceed $250,000, will be made for purchase of special laboratory equipment.

• Argonne National Laboratory's International School of Nuclear Science and Engineering has 62 students, 50 of them from 21 foreign nations, enrolled in its sixth session. This enrollment brings to 355 the number of scientists and engineers who have pursued the course in unclassified reactor technology.

Courses

Five training courses in **radiological health and air and water pollution** have been scheduled for January–March at the Robert A. Taft Sanitary Engineering Center, a U. S. Public Health laboratory in Cincinnati. The schedule includes: Jan. 13–24, basic radiological health; Jan. 27–31, environmental health aspects of nuclear reactor operations; Feb. 17–21, detection and control of radioactive pollutants in air; Feb. 24–28, detection and control of radioactive pollutants in water; and March 17–28, sanitary engineering aspects of nuclear energy. No tuition fee will be charged.

Training in radioisotope techniques will be offered to Latin American scientists at the newly-established Radioisotope Techniques Training Center at the Univ. of Puerto Rico on the following dates: Jan. 6–31, 1958; March 3–28; and May 5–30.

A seminar on fusion is being given at Stevens Institute of Technology, Hoboken, N. J., each Tuesday at 3 p.m. and will continue through May. Subjects to be discussed during December and January at this, perhaps the first nonclassified fusion colloquium, include: experiments on the pinch effect, collective coordinates for plasmas, production of strong magnetic fields, and force-free magnetic fields. There is no fee for the seminar.

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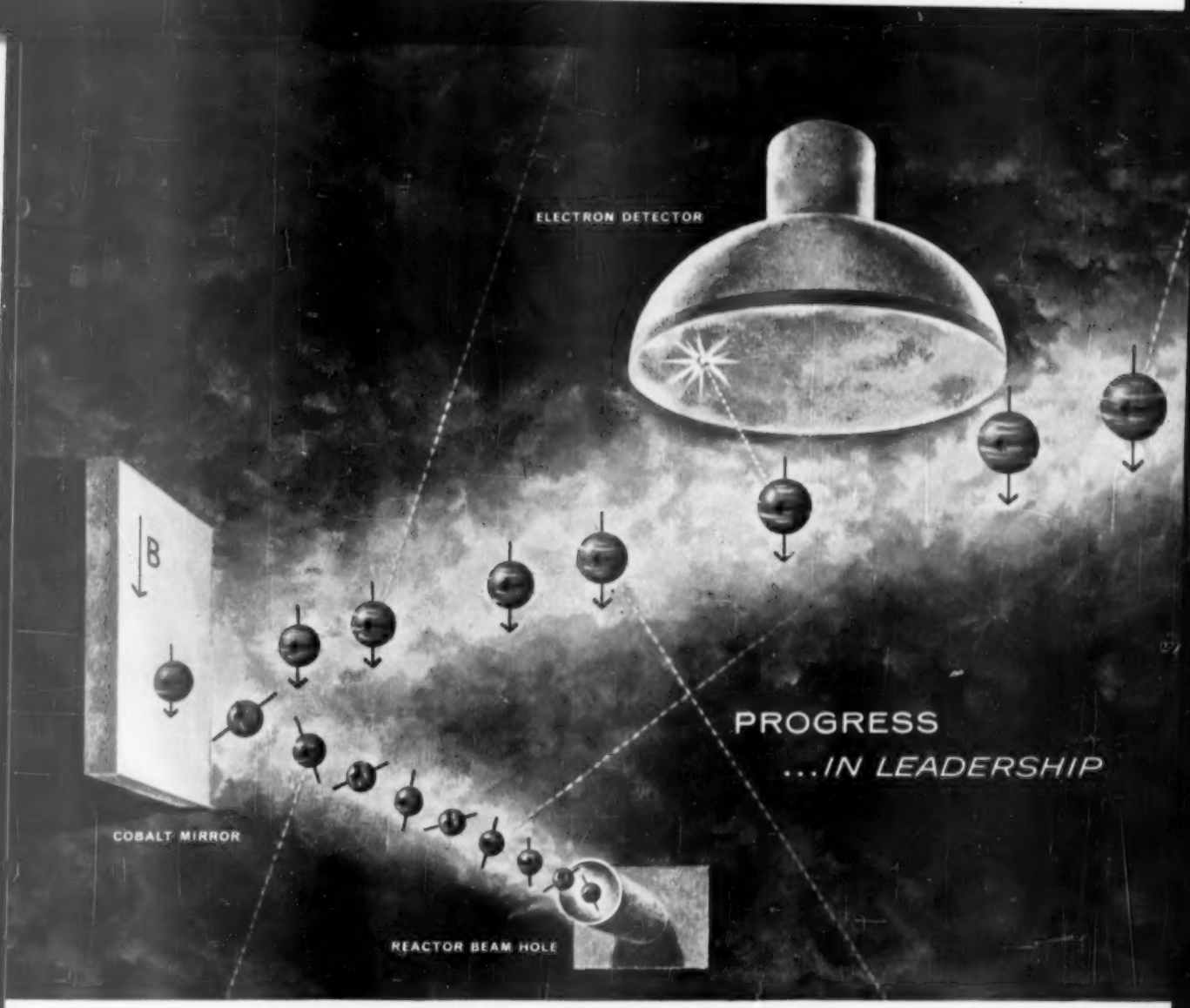

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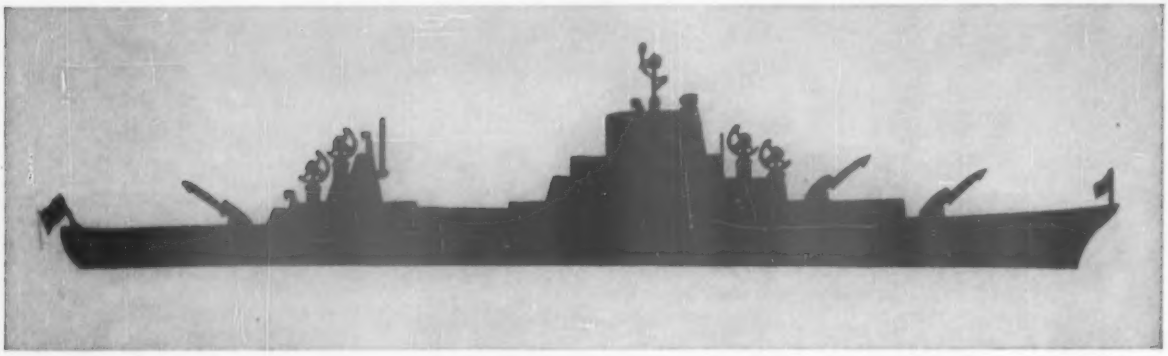

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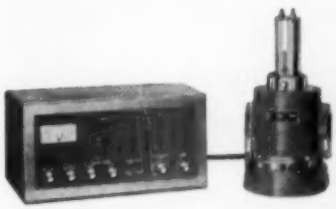

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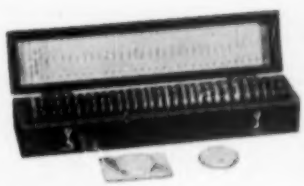

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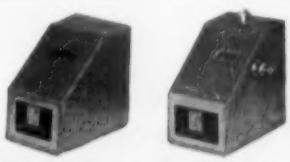

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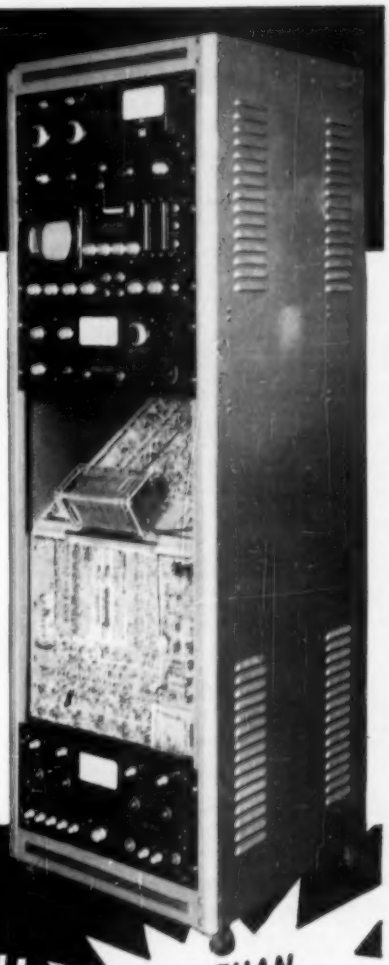

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INDEX to VOLUME 15, January—December, 1957

How to Use the Index

As in the case of previous NUCLEONICS indexes, we have introduced a number of changes in this year's index headings.

Since more 1957 articles are devoted to reactor topics than ever before, many are now found under such headings as "Components, reactor" and "Fuels, reactor." On the other hand, descriptions of complete reactors are indexed under "Nuclear reactors, power," "Nuclear reactors, propulsion," "Nuclear reactors, test," etc. Companies are listed by name under the heading "Industry."

Other minor changes have been made. However, any difficulty should be precluded by liberal cross-referencing.

The chemical elements are indexed by the name of the particular element, e.g., cobalt, thorium, uranium, zirconium.

Particular countries are indexed individually, e.g., Britain, France and Canada, as are various organizations, e.g., Atomic Energy Commission, though not individual companies, which are found under "Industry."

Feature articles are for the most part indexed under three or more headings, depending on the nature of the article. News items and shorter departmental articles (CS, NE, AR, etc.*) are indexed under fewer headings since they are usually more specific in subject matter.

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